TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TRANSIENTS

LIST OF TRANSIENTS (CONT'D)

5.7 Accidents

Exercises

This page intentionally blank

5.0 Westinghouse Four-Loop Design Transients

Learning Objectives:

1. Given a set of transient curves and Table 5-1, demonstrate an understanding of plant characteristics and control, protection, and safeguards systems by:
   a. Explaining why the parameter values are trending as shown at selected numbered portions of the curves,
   b. Explaining plant effects caused by parameters reaching certain values at selected numbered points, and
   c. Explaining the cause(s) of the reactor trip and/or engineered safety features (ESF) actuation, if either occurs.

This page intentionally blank

5.1 Introduction

The transient curves contained in this chapter were compiled and analyzed by staff members of the NRC's Technical Training Center (TTC). They were produced from the dynamic responses of the Trojan (a Westinghouse four-loop reactor plant) training simulator. Specific parameter responses of the simulator were recorded by a data acquisition program and then graphed with a graphics program.

The instructor explanations provided in class for these curves are the results of analysis by the TTC staff during the actual simulator "runs" and during subsequent staff seminars. For each transient, the sequence of numbered points has been established to aid the instructor's classroom presentation.

Caution is advised when trying to apply these simulator curves to any operating plant. Even relatively minor changes in setpoints, capacities, or plant configurations could cause significant differences in indicated responses.

During analysis and study of the curves, the student should concentrate on explaining the changes in various parameters caused by the initiating event and by the subsequent operation of control, protection, and safeguards systems. When explaining a numbered point, the student should always try to relate "cause" and "effect" (e.g., pressurizer level is increasing because the reactor coolant system [RCS] average temperature is increasing, and the coolant is expanding into the pressurizer). Do not place too much emphasis on an isolated portion of or a minor deviation in the graph of a particular parameter unless it is associated with a numbered point. Generally, a numbered point will bracket a portion of a curve, indicating that the student should try to explain why a parameter is trending or changing in the bracketed area. If a numbered point is associated with a reactor trip or engineered safety features actuation, the student should attempt to explain not only that the protective action has occurred but also what reactor trip signal or ESF actuation signal is present.

The following general notes are applicable to all transients unless other information is provided:

1. Pressurizer pressure is from one of the four pressurizer pressure instruments. In a few transients, wide-range RCS pressure from one of the pressure detectors on the residual heat removal (RHR) system suction line is also provided.
2. Bank D rod position is from the demanded position indication system (shows every step change).
3. Nuclear power is from one of the four excore nuclear instruments.
4. Generator load is in electrical MW.
5. Average RCS temperature (T_{avg}) is the T_{avg} from one of the four coolant loops, derived from the narrow-range resistance temperature detectors (RTDs) in the bypass manifold. The programmed T_{avg} for a particular turbine load (T_{ref}) is a function of turbine impulse pressure.
6. Pressurizer level is from one of the three pressurizer level detectors.

7. Charging flow is from the flow transmitter downstream of the charging pumps and includes flow supplied to both the normal charging line and to reactor coolant pump seal injection.

8. Steam dump demand is the ouput of either the loss-of-load, the turbine trip, or the steam pressure controller, whichever is in service.

9. Steam flow (W_s) is the flow in one of the four main steam lines but is indicative of total steam flow.

10. Feedwater flow (W_f) is the flow supplied to one of the four steam generators but is indicative of total feedwater flow.

11. Steam generator level is from one of the three narrow-range level detectors on one of the four steam generators but is indicative of the level in any steam generator.

12. Steam pressure (P_{stm}) is from one of the three pressure detectors on one of the four main steam lines but is indicative of the pressure in any steam line.

13. Additional parameters are monitored and graphed if they are pertinent to the transient analysis.

14. When a transient is caused by a control system response to an instrument failure, the output of a redundant instrument is graphed to display the actual changes in the parameter of interest.

15. Initial plant conditions not available from the transient curves are given by the instructor during the introduction to the transient and listed in a box adjacent to the transient curves. For transients used on the final exam, the initial conditions are given as part of the problem statements.

5.2 Transient Analysis

The following sections discuss various aspects of transient analysis.

5.2.1 Energy Equilibrium

Transient analysis begins with an examination of the stored energy of the reactor coolant. As shown in Figure 5-1, the internal energy of the reactor coolant is dependent on two factors, the energy input from the core and the energy removal by the secondary system (steam generators). If the energy input equals the energy removal, then the internal energy of the reactor coolant is not changing. Therefore, the average coolant temperature is stable. However, if an upset in the energy equilibrium occurs, then the internal energy of the reactor coolant changes, resulting in a change in coolant temperature. When a change in coolant temperature occurs, the density of the reactor coolant changes. The changes in temperature and density affect several of the parameters that are shown in the transient curves of this chapter.

Assume that with an initial equilibrium between energy production and energy removal, a transient occurs that results in a reduction in the rate of energy removal (e.g., a turbine load reduction). Since the rate of energy production (reactor power) can not immediately drop, the internal energy of the reactor coolant increases, and the average coolant temperature increases. When the coolant temperature increases, the density of the coolant decreases. This decrease in density results in an increase in the volume of the reactor coolant, causing an insurge into the pressurizer and an increase in pressurizer level. The pressurizer level insurge compresses the steam bubble, and pressurizer pressure increases.

Now consider an increase in the rate of energy removal by the secondary system (e.g., a turbine load increase) from equilibrium conditions. Initially, the rate of energy removal from the reactor coolant exceeds the rate of energy production by the reactor, the internal energy of the reactor coolant decreases, and the average coolant temperature decreases. When the coolant temperature decreases, the density of the coolant increases. The immediate consequence of an increase in coolant density is an outsurge from the pressurizer and a corresponding decrease in pressurizer level. When the pressurizer level decreases, the volume of the steam bubble increases. The expanding steam bubble results in a decrease from the initial pressurizer pressure.

In each of the examples discussed above, the reactor coolant temperature and density and the pressurizer level and pressure change as a result of a change from an initial equilibrium between the energy input to and energy removal from the reactor coolant.

A change in the stored energy of the reactor coolant can be identified by comparing the reactor power and the steam demand on the steam generators. Generally, if the turbine load is less than the reactor power, then the average coolant temperature is increasing, and conversely, if the turbine load is greater than the reactor power, then the average coolant temperature is decreasing. Any time the turbine is not in service or an additional steam demand from steam dump operation or a steam break is present, a comparison of steam flow and reactor power leads to the same conclusions. Once the direction of the energy mismatch is known, the changes in coolant temperature and in pressurizer level and pressure can be explained.

The two examples in the previous discussion are representative of two types of transients. In the first type, reactor power exceeds the rate of energy removal by the secondary; if the mismatch is extreme, the transient is referred to as an overheating event. This type of transient includes turbine trips, load rejections, and normal power decreases. In the second type, the rate of energy removal by the secondary exceeds reactor power; if the mismatch is extreme, the transient is referred to as an overcooling or excessive heat transfer event. Examples of this type of transient are normal power increases, steam dump operation, steam generator power-operated relief valve (PORV) openings, turbine valve failures, and steam line breaks.

In addition to determining the direction and magnitude of the energy input/energy removal mismatch, the student must analyze the responses of the control systems. If nuclear power exceeds turbine load, T_{avg} increases. If T_{avg} increases above T_{ref}, then the control rods are inserted by the rod control system (assuming automatic operation). Also, the pressurizer level increases. If the increase in level exceeds the increase in the pressurizer level setpoint, the pressurizer level control system decreases charging flow. The accompanying increase in pressurizer pressure is compared to the pressure setpoint in the pressurizer pressure control system. The control system reduces the output of the proportional heaters and, if the pressure error is large enough, opens the spray valves. Finally, if the increase in pressurizer pressure is large enough, the pressurizer PORVs open. The rod control system and the pressurizer level and pressure control systems will react in similar but opposite fashions to a transient in which turbine load exceeds nuclear power.

5.2.2 Reactivity Balance

Transient analysis also involves an examination of the reactivity balance. The transients in this section can involve changes in fuel temperature, moderator temperature, and control rod position, any of which can add positive or negative reactivity to an initial state of equilibrium reactivity ($\rho = 0$). For the transients of this section, the fuel and moderator temperature coefficients of reactivity are always negative. No transient time span is long enough for changes in fission product (poison) concentrations to significantly affect reactivity, and no transient involves an operator-controlled change in boron concentration. If the transient terminates at a new steady-state endpoint without a plant trip, the positive reactivity added by one source must be completely balanced by the negative reactivity added by another.

During a normal load change, reactivity will be added by the power defect and compensated by a change in control rod position. The power defect (the power coefficient integrated over a power change) accounts for the change in reactivity associated with the changes in fuel temperature and moderator temperature, with the moderator temperature assumed to be maintained at programmed values. When the operator changes the turbine load at the turbine electrohydraulic control (EHC) station, the resulting primary-to-secondary mismatch causes the average coolant temperature to initially increase or decrease. The rod control system (if in automatic) responds to the T_{avg}/T_{ref} error and the power mismatch associated with the load change by inserting or withdrawing rods. When the new steady state has been reached at the end of the load change, the reactivity balance ($\rho = 0$) is restored, with the reactivity associated with the power defect completely balanced by the reactivity added by the change in control rod position.

As an example, consider a turbine load reduction with the rod control system in automatic. Initially, the drop in load relative to the unchanged nuclear power causes the average reactor coolant temperature to increase, and the temperature and power mismatch circuits of the rod control system call for control rod insertion. The control rod insertion suppresses nuclear power and drives down T_{avg} to match the decreasing T_{ref}. Meanwhile, the fuel temperature is decreasing with the decrease in nuclear power. When the load change is complete, the primary power again equals the secondary load, and the positive reactivity addition associated with the power defect (both fuel and moderator temperatures are lower at the transient endpoint) is completely balanced by the negative reactivity added by the control rod insertion.

Next, consider the load reduction with the rod control system in manual. The primary-to-secondary power mismatch increases the coolant temperature and thereby adds negative reactivity. The negative reactivity addition decreases reactor power. The decrease in reactor power adds positive reactivity via the fuel temperature coefficient (the fuel temperature is decreasing), resulting in a dampening of the power decrease. As long as the rate of reactor energy production is greater than the rate of energy removal by the turbine, the coolant temperature continues to rise. The transient is terminated when the rate of energy input to the coolant by the reactor exactly matches the rate of energy removal by the secondary system, and the positive reactivity addition associated with the decrease in fuel temperature exactly matches the negative reactivity addition associated with the increase in coolant temperature. The endpoint conditions are equal values of reactor and secondary power and a T_{avg} that is higher than that at the start of the transient.

The examples discussed above involve changes initiated by the secondary plant. However, transients can be initiated in the primary system. An uncontrolled rod withdrawal and a dropped rod are two examples. However, the considerations of any existing energy mismatch, control system actions, and the effects of reactivity coefficients remain applicable. For the transients in this section, the moderator and fuel temperature coefficients and the reactivity changes associated with rod motion account for the changes in reactor power. In actual plant operation, long-term changes in the concentrations of fission product poisons and operator-controlled changes in the boron concentration must also be considered.

5.2.3 Steam Generators

Another consideration in the analyses of transients involves the changes that occur in steam generator level and pressure. The initial changes in steam generator level that are caused by changes in steam flow from the steam generator are called "shrink" and "swell." It is important to recognize that such a change in indicated level is not caused by a change in steam generator inventory. Recall that SG level is measured in the downcomer, not the tube bundle region. Water flows from the downcomer to the tube bundle region by gravity.

The static pressure of the water column in the downcomer must be greater than the static pressure of the water/steam mixture in the tube bundle region to induce flow from the downcomer to the tube bundle region. This pressure difference requires a difference in the water levels in the two regions. (The level in the tube bundle region is an effective level because the fluid volume there is partially steam.) In a static condition (no flow), these two levels would be equal. As steam flow from the turbine increases (as with a turbine load increase), the rate of steam formation in the tube bundle region increases. The increased steam fraction of the tube bundle fluid volume increases the resistance to flow (head loss) between the downcomer and the tube bundle region; the increased flow resistance requires a greater difference in the effective levels of those two areas. The result is a rise in the downcomer water level. Conversely, for a decrease in steam flow from the SG, the flow resistance between the downcomer and the tube bundle region drops, resulting in a reduced downcomer level.

In summary, with an increase in steam flow, indicated level inherently rises. This is called swell. With a drop in steam flow, indicated level inherently drops. This is called shrink. The change in indicated level is due to a change in the flow resistance between the downcomer and the tube bundle region. The change in indicated level is not due to a change in steam generator inventory.

Following the initial change in level, the steam generator water level control system (SGWLCS) returns the level to the normal programmed value through a change in feedwater flow.

For reasons not fully understood, the wide range level indication is generally not affected by shrink and swell.

5.2.4 Instrument Failures

A knowledge of control system functions and actions that are taken at particular setpoints is necessary to analyze instrument failure transients. A failure of an instrument which feeds an input to a control system can be analyzed by asking the following questions:

1. What is the function of the control system?
2. What actions does the control system take to accomplish its function?
3. What actions are taken if the actual value of the parameter is above or below the setpoint value?

In short, if the output of a failed instrument is supplied to a control system, the student should determine the response of the control system and how the controlled component changes plant conditions.

As an illustration of this technique, consider the case of a controlling steam generator level transmitter failing low. The inaccurate level is provided to the SGWLCS; the function of the SGWLCS is to maintain the steam generator level at the setpoint value. The first question in the above list is now answered. The SGWLCS controls the steam generator level at setpoint by controlling the position of the main feedwater regulating valve. The second question is now answered. Finally, if the steam generator level is low, the feedwater regulating valve opens further to increase the level in the steam generator. Since the SGWLCS has no way of "knowing" that it has a faulty input, this response occurs even with an initially normal steam generator level. Now consider the resulting effects. Feedwater flow now exceeds steam flow, and the steam generator level increases. This example illustrates the basic questions to be kept in mind for analyses of transients initiated by instrument failures.

5.2.5 Accidents

Analyses of accidents generally involve the trends in primary and secondary levels and pressures and the responses of plant safeguards systems. In the case of a loss of coolant accident (LOCA), the pressurizer pressure and level drop, but the steam generator pressures and levels are largely unaffected. Since a steam generator tube rupture (SGTR) is a special form of LOCA, the primary conditions will change similarly during an SGTR, while the level in the affected steam generator increases with the influx of reactor coolant through the rupture. Steam line breaks can be grouped into breaks upstream of the main steam isolation valves (MSIVs) and downstream of the MSIVs. During a break upstream of the isolation valves, the steam pressure in the affected steam generator decreases more rapidly than the pressures in the unaffected steam generators. Following isolation of the faulted steam generator by its check valve, the pressures in the intact steam generators should recover, while the affected steam generator blows down to atmospheric pressure. A break downstream of the MSIVs results in equal pressure drops in all steam generators, which are terminated by MSIV closure. Of course, the overcooling of the reactor coolant caused by a steam break also lowers pressurizer pressure and level.

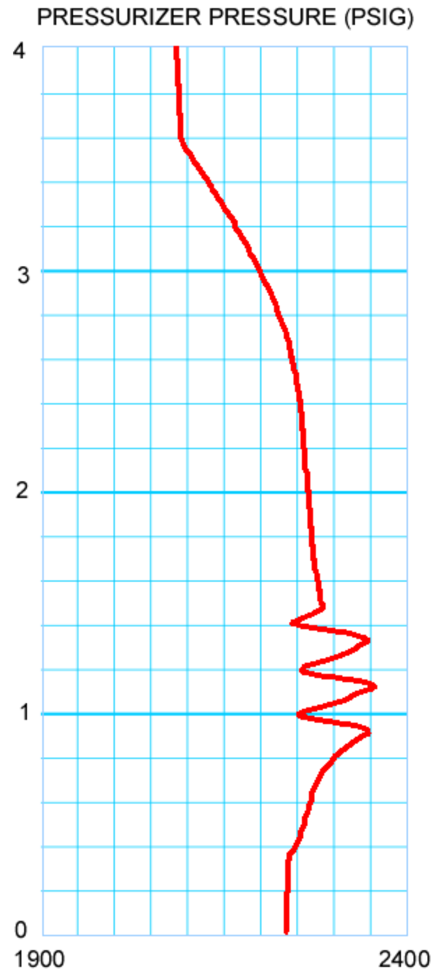
For any accident, an SI actuation is indicated by the change in charging flow upon the isolation of normal charging and the initiation of high head injection, and by the change in feedwater flow upon the isolation of main feedwater and the initiation of the auxiliary feedwater system. During steam line breaks and some small LOCAs, high head injection eventually reverses the drop in pressurizer level caused by overcooling of the reactor coolant or by inventory loss. For some transients, plots of high, intermediate, and low head injection are provided to illustrate the responses of the emergency core cooling systems to an SI actuation and plant conditions, and plots of containment pressure are provided to illustrate the progress of the accident and the response of containment pressure suppression systems.

In an actual reactor plant, indications of accidents would include the responses of radiation detectors. Elevated containment radiation levels would result from a LOCA, and higher secondary radiation indications would result from a primary-to-secondary leak. No radiation indications are included as part of the transient curves provided in this manual.

5.2.6 Trend Format

The trends in this chapter present data in a way that may be unfamiliar to the student. Traditionally, parameter change vs. time is displayed in a format where time is the parameter associated with the horizontal (X) axis. However, in the trends for this course, time is on the vertical (Y) axis, and the parameter value of interest, which is graphed as a function of time, is on the horizontal (X) axis. This convention was established by the control room chart recorders initially installed in the units. The paper strip moved down as the pen recorded the value, leaving the most recent value at the top of the chart. Since operators were accustomed to this format, this convention was retained in digital replacements. We present the data in the same format as that displayed on control room trend recorders.

On the graph to the right, for instance, pressurizer pressure is plotted versus time over a four-minute interval. The scale for pressurizer pressure, as shown at the bottom of the graph, is 1900 – 2400 psig.

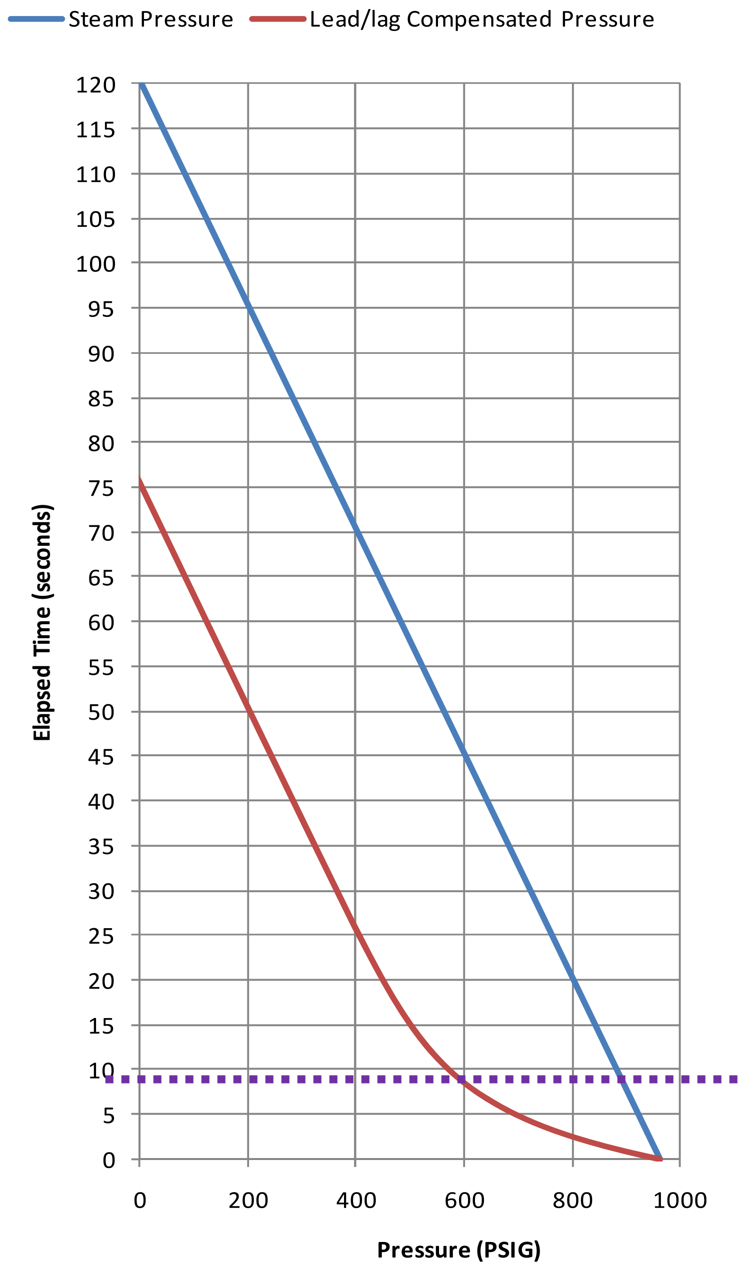


5.3 Signal conditioning

Some process control circuits contain components which modify the signal measurement.

5.3.1 Lead/Lag Compensation

The figure to the right illustrates the effect of a lead/lag circuit where the lead time constant is 50 seconds, (τ_1), and the lag time constant is 5 seconds, (τ_2). These are the time constants associated with the modified steam pressure signal which is an input to the low steam pressure bistables in the reactor protection system. The associated low steam pressure bistable does not "see" the measured steam pressure; it only "sees" the output of the lead/lag unit. In this example, if steam pressure drops at a constant rate, the output of the lead/lag function eventually resolves to a linear function where the output reaches a specific value 45 seconds (lead minus lag) before the actual parameter reaches that value. For example, the lead/lag output reaches 400 psig at time 25 seconds, and the actual parameter reaches 400 psig 45 seconds later (70 seconds). It takes about 5 lag time constants to resolve to a linear function.

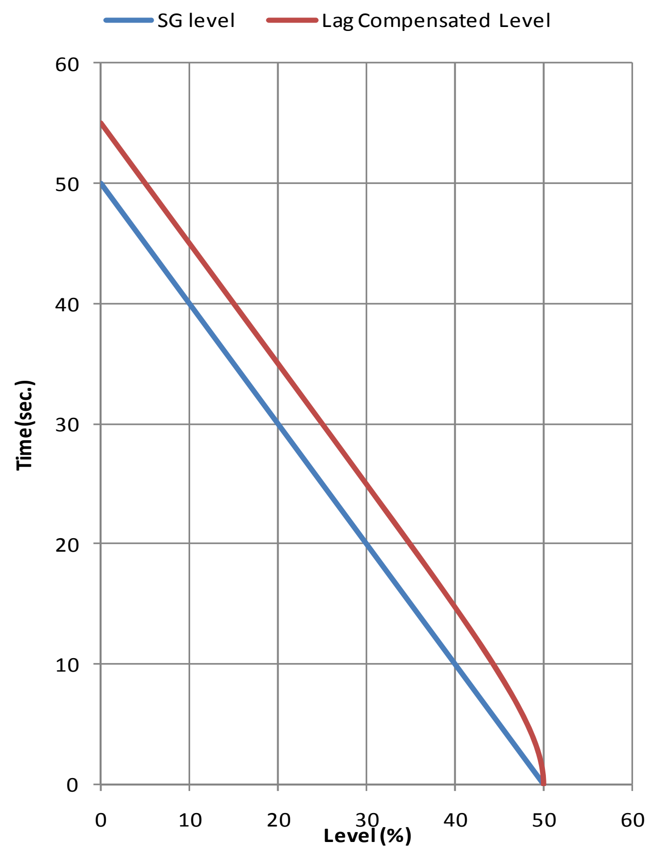


If steam pressure follows this trend, the low pressure bistable, with a trip setpoint of 600 psig, trips at an indicated pressure of ~875 psig.

The lead/lag function is represented on plant instrument diagrams by $\frac{1+\tau_1 s}{1+\tau_2 s}$.

5.3.2 Lag Compensation

The figure to the right illustrates the effect of a lag circuit, where the lag constant is 5 seconds, (τ_1), as in the level input to the steam generator level control system. In this example, if steam generator level drops at a constant rate, the output of the lag function eventually resolves to a linear function where the output reaches a specific value 5 seconds after the actual parameter reaches that value.

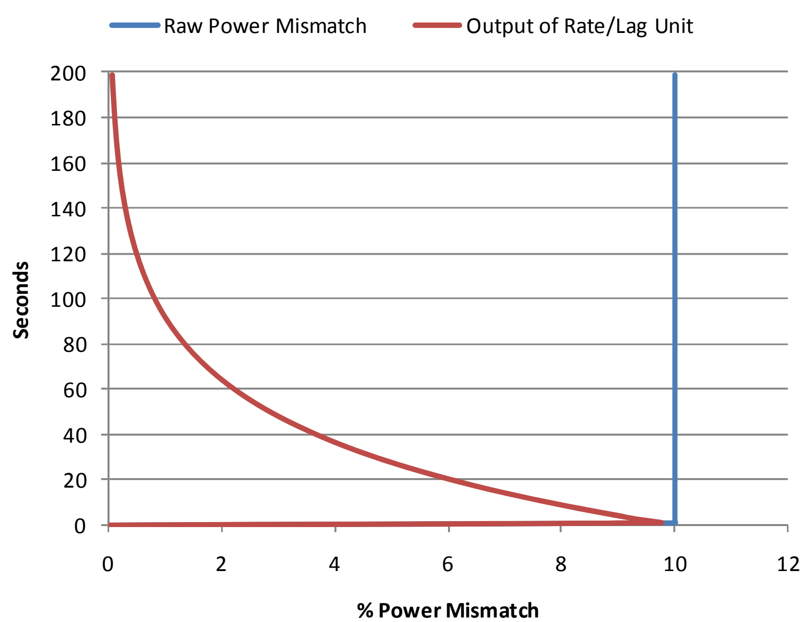


When steam generator level begins to change, the steam generator level control system does not immediately "see" this change. This prevents an inappropriate response to shrink and swell.

The lag function is represented on plant instrument diagrams by $\frac{1}{1+\tau_1 s}$.

5.3.3 Rate/Lag Compensation

The figure to the right illustrates the effect of a rate/lag circuit where the rate time constant, (τ_1), is 40 seconds, and the lag time constant, (τ_2), is 40 seconds, as in the rod control power mismatch unit. The rate/lag unit output is only generated in response to changes in the input. The figure shows the effect of a step change in the power mismatch, from 0% to 10% at time zero. This results in a step change in the rate-lagged signal. When the power mismatch stays constant, the output of the rate/lag unit decays to zero in about 5 lag time constants. The rate/lag function is represented on plant instrument diagrams by $\frac{\tau_1 s}{1+\tau_2 s}$.



5.4 Parameter Behavior During Transients

The following descriptions of parameter behavior during transients are provided in the order with which the graphs of the parameters are presented.

5.4.1 Pressurizer Pressure

1. Pressurizer pressure is affected by components controlled by the pressurizer pressure control system. This is particularly evident during transients involving the failure of the controlling pressure channel.
2. A rapid change in pressurizer level can have such a large effect on the dimensions of the pressurizer steam bubble and, as a result, on pressurizer pressure that the pressurizer pressure control system cannot immediately restore pressure to setpoint.
3. This parameter is an input into the OTΔT trip and turbine runback setpoint calculations and can cause the setpoints to increase or decrease. Evidence of a turbine runback can be seen on the generator load plot.

5.4.2 Bank D Rod Position

1. Bank D rod position is affected by the power mismatch and temperature mismatch inputs to the rod control system.
2. It is possible for the power mismatch circuit output to be equal and opposite to the temperature mismatch circuit output. This condition results in no rod motion, even though a T_{ref} - T_{avg} difference exists.
3. The failure of an the input to the power mismatch circuit causes rapid rod motion initially due to the high rate of change of nuclear power relative to turbine load; the output of the power mismatch circuit then decays exponentially, allowing any existing temperature mismatch to gradually increase its impact on rod control.
4. A step drop in bank D rod position to 0 steps is indicative of a reactor trip.

5.4.3 Nuclear Power

Nuclear power responds to reactivity effects associated with fuel temperature, moderator temperature, and control rod position. No transient time span is long enough for changes in fission product (poison) concentrations to significantly affect reactivity. No transient involves an operator-controlled change in boron concentration; changes in the coolant boron concentration occur only during transients involving significant injection of the refueling water storage tank contents.

5.4.4 Generator Load

1. During power level changes, the change in generator load is usually the initiating event. A load change can be input gradually by the operator with the selection of a new demanded load and loading rate or rapidly via operation of the control valve position limiter.
2. The Trojan GE turbine EHC system generates a demanded control valve position for a given demanded load and does not incorporate MW or impulse pressure feedback. Thus, once the control valves reach their demanded positions, they will not respond to load changes if the demanded load remains unchanged. The

throttle pressure compensation circuit will attempt to maintain constant steam flow over the range of no-load (~1020 psig) to full-load (~750 psig) throttle pressure. Throttle pressure changes inside this band only produce minor changes in generator load. Throttle pressure changes outside this band cause significant changes in generator load. For example, RCS temperature changes initiated from 50% power do not cause significant changes in generator load, since throttle pressure compensation is in the center of its operating band. Cooldown transients initiated from rated thermal power will cause generator load to drop, since throttle pressure drops below the range of throttle pressure compensation.

3. The Trojan GE EHC system includes an initial pressure limiter which closes the control valves when throttle pressure drops below 90% of the throttle pressure for rated power. The response of this EHC system feature is evident in certain generator load reductions in some transients.

4. A turbine runback is indicated by an abrupt change in load to a new lower value.

5. A step drop in generator load to 0 MW is indicative of a turbine trip.

5.4.5 T_{ref}/T_{avg}

1. Since T_{ref} varies linearly with impulse pressure, it reflects changes in generator load.

2. T_{avg} is generated from the hot-leg and cold-leg temperatures (T_H and T_c) measured in the resistance temperature detector (RTD) bypass manifolds. This arrangement contributes to the inherent delay between the time a T_{avg} change occurs and the time the T_{avg} change is indicated. The delay involved is due to the coolant loop transport time and the time required for coolant to flow through the bypass manifold to the narrow-range RTD locations. Therefore, during a rapid transient the pressurizer level provides a better initial indication of a coolant temperature change (see section 5.3.6 below).

3. T_{avg} is a reflection of the balance between the rate of energy production in the primary and the rate of energy removal by the secondary. If the two are equal, T_{avg} will remain constant. Any imbalance, whether initiated in the primary or secondary, causes a change in T_{avg}.

5.4.6 Pressurizer Level

1. A change in pressurizer level is often a direct reflection of a change in reactor coolant density and thus provides an indication of a primary temperature change.

2. A decrease in pressurizer level can be indicative of a loss of coolant inventory.

3. A somewhat small but visible change in pressurizer level can result from a change in coolant density associated with a moderately large pressure change.

5.4.7 Charging Flow

1. Generally, charging flow varies with the position of charging flow control valve FCV-121, which responds to the output of the pressurizer level control system (all transients begin with charging flow supplied by one centrifugal charging pump). Charging flow increases when the pressurizer level is less than the level setpoint

and decreases when the level is greater than the setpoint. Often during a transient the pressurizer level and the level setpoint (a function of auctioneered high T_{avg}) are changing in the same direction simultaneously but not in step, so that charging flow undergoes "swings" in which it first increases and then decreases, or vice versa.

2. An SI actuation signal causes a characteristic perturbation in charging flow during which the second centrifugal charging pump starts, the normal charging line isolates, and charging flow becomes seal injection only. This perturbation appears on the charging flow plot as a "zigzag." The steady-state charging flow after an SI actuation depends on the RCS pressure and the position of FCV-121, which continues to modulate in response to pressurizer level control system commands.

5.4.8 Steam Dump Demand

During power operation a steam dump demand indication reflects a T_{avg} - T_{ref} difference of greater than 5°F (the loss-of-load controller is in service). The T_{avg} signal is rate compensated, so if T_{avg} is rising quickly, the demand signal may be generated at an indicated ΔT of < 5°F. Following a turbine trip, an existing demand indicates that T_{avg} exceeds the no-load T_{avg} (the turbine trip controller is in service). During plant heatups and startups, an existing demand indicates that steam pressure exceeds the no-load steam pressure setpoint of 1092 psig. A demand indication does not necessarily mean that the steam dumps are opening; an arming signal must also be present. The best confirmation of steam dump operation is a change in steam flow. When steam dump demand is indicated, an increase in steam flow indicates that dump valves are open.

5.4.9 Steam Flow

Steam flow responds to changes in turbine control valve position, steam generator PORV operation, steam generator safety valve operation, and steam dump operation.

5.4.10 Feedwater Flow

1. Feedwater flow is governed by the position of the main feedwater regulating valve, which is controlled by the SGWLCS.
2. At the outset of a transient, the change in feedwater flow is governed by the feed flow/steam flow mismatch. As the transient progresses and the level error has a chance to build, the level error signal will dominate feedwater flow changes.
3. Feedwater flow often undergoes many oscillations during a transient. Large swings in feed flow correspond to significant changes in main feed regulating valve position; small-amplitude fluctuations in feed flow may be considered as normal steady-state operation.
4. The feedwater flow indication following the isolation of main feedwater reflects auxiliary feedwater addition to the steam generator. In the control room, main feedwater flow and auxiliary feedwater flow are indicated on separate meters.

5.4.11 Steam Generator Level

1. A rapid change in steam demand causes a shrink or swell to occur (see section 5.2.3). Wide range steam generator level, which is displayed for some transients, is generally not affected by shrink or swell.

2. A change in the reactor coolant temperature, especially a decrease, can result in a change in the secondary temperature of the steam generators and changes in steam density and steam generator level.

3. Following the isolation of main feedwater, level is affected by auxiliary feedwater addition.

5.4.12 Steam Pressure

1. In general, steam pressure increases with a load decrease and decreases with a load increase.

2. Steam pressure can be affected by a change in T_{avg} if the change is large enough to affect the conditions governing primary-to-secondary heat transfer (see section 5.3.11).

3. A rapid drop in steam pressure can reflect operation of the steam generator PORVs and safety valves and steam line breaks.

TABLE 5-1 TRANSIENT INFORMATION

I. Setpoints

A. Reactor Coolant Temperature (°F)

564	Low T_{avg}
557 - 584.7	T_{avg} program from 0% to 100% power
553	Low-low T_{avg} (P-12)

B. Pressurizer Level (% level)

92	High level reactor trip
25 - 61.5	Level program from 0% to 100% power
17	Low level heater cutoff and letdown isolation

C. Pressurizer Pressure (psig)

2485	Code safety valves open
2385	High pressure reactor trip
2335	PORVs open
2235	Nominal operating pressure
1915	Low pressure SI block permissive (P-11)
1865	Low pressure reactor trip
1807	Low pressure SI actuation

D. Steam Generator Level (% level)

69	High level turbine trip, feedwater isolation, trip of main feed pumps (P-14)
44	Program level from 20% to 100% power
33 - 44	Level program from 0% to 20% power
25.5	Low level reactor trip (with steam flow > feed flow by 1.51×10^6 lbm/hr)
11.5	Low-low level reactor trip, AFW actuation

E. Steam Dump System Controller Inputs (°F)

5 - 16.4	Generates 0 - 100% output from loss-of-load controller
0 - 27.7	Generates 0 - 100% output from turbine trip controller

F. Nuclear Instrumentation

1. Source Range (cps)

   10^5 High flux reactor trip

2. Intermediate Range

25% current equivalent	High flux reactor trip
20% current equivalent	High flux rod stop
10^{-10} amps	Source range block permissive (P-6)

3. Power Range (% power)

109	High flux, high setpoint reactor trip
103	High power rod stop
39	Loss of loop flow permissive (P-8)
25	High flux, low setpoint reactor trip
10	Nuclear at-power block permissive (P-10)
+5 (w/ 2-sec time constant)	Positive high flux rate reactor trip
-5 (w/ 2-sec time constant)	Negative high flux rate reactor trip

G. Main Steam Pressure (psig)

1170 -1230	Range of code safety valve lift setpoints
1125	Atmospheric relief valve lift setpoint
600	Low steam pressure SI actuation (with high steam flow)

H. ESF Actuation Signals

Refer to Technical Specification Table 3.3.2-1

II. Significant Parameters (Typical Values)

A. Reactivity Values

1. Moderator Temperature Coefficient (no-load)

   BOL: -4 pcm/°F (1500 ppm boron)
   EOL: -26 pcm/°F (0 ppm boron)

2. Doppler-Only Power Coefficient

   BOL: -13 pcm/% power
   EOL: -11 pcm/% power

3. Power Defect at 100% power

   BOL: -1500 pcm
   EOL: -2400 pcm

4. Control Rod Worths

   Bank: 1000 pcm
   Individual: 150 pcm
   Differential worth: 4 to 12 pcm/step

5. Xenon Reactivity (BOL)

   Equilibrium at 100% power: -2741 pcm
   Peak following reactor trip: -5200 pcm

6. Reactor Makeup Parameters

   Boric acid worth: 8 pcm/ppm (BOL)
   Maximum dilution rate: 120 gpm
   Maximum boration rate: 40 gpm (4 weight % boric acid)
   Automatic makeup rate: 80 gpm total blended flow

B. System and Component Parameters

1. RCS

   Range of ΔT from 0% to 100% power: 0 - 59°F

2. Pressurizer

   1% change in level per °F change in T_{avg}
   130 gal per % level
   10 psi change in pressure per % change in level
   10 psi change in pressure per °F change in T_{avg}

3. Main Steam System

   No-load pressure (corresponds to T_{avg} of 557°F): 1092 psig
   Full-load pressure: 792 psig
   Steam flow per generator (100% power): 3.77×10^6 lbm/hr
   Total steam flow (100% power): 15.07×10^6 lbm/hr

4. ECCS Maximum Pressures for Injection (psig)

   2670 HPI pumps
   1520 SI pumps
   650 Cold-leg accumulators
   200 RHR pumps

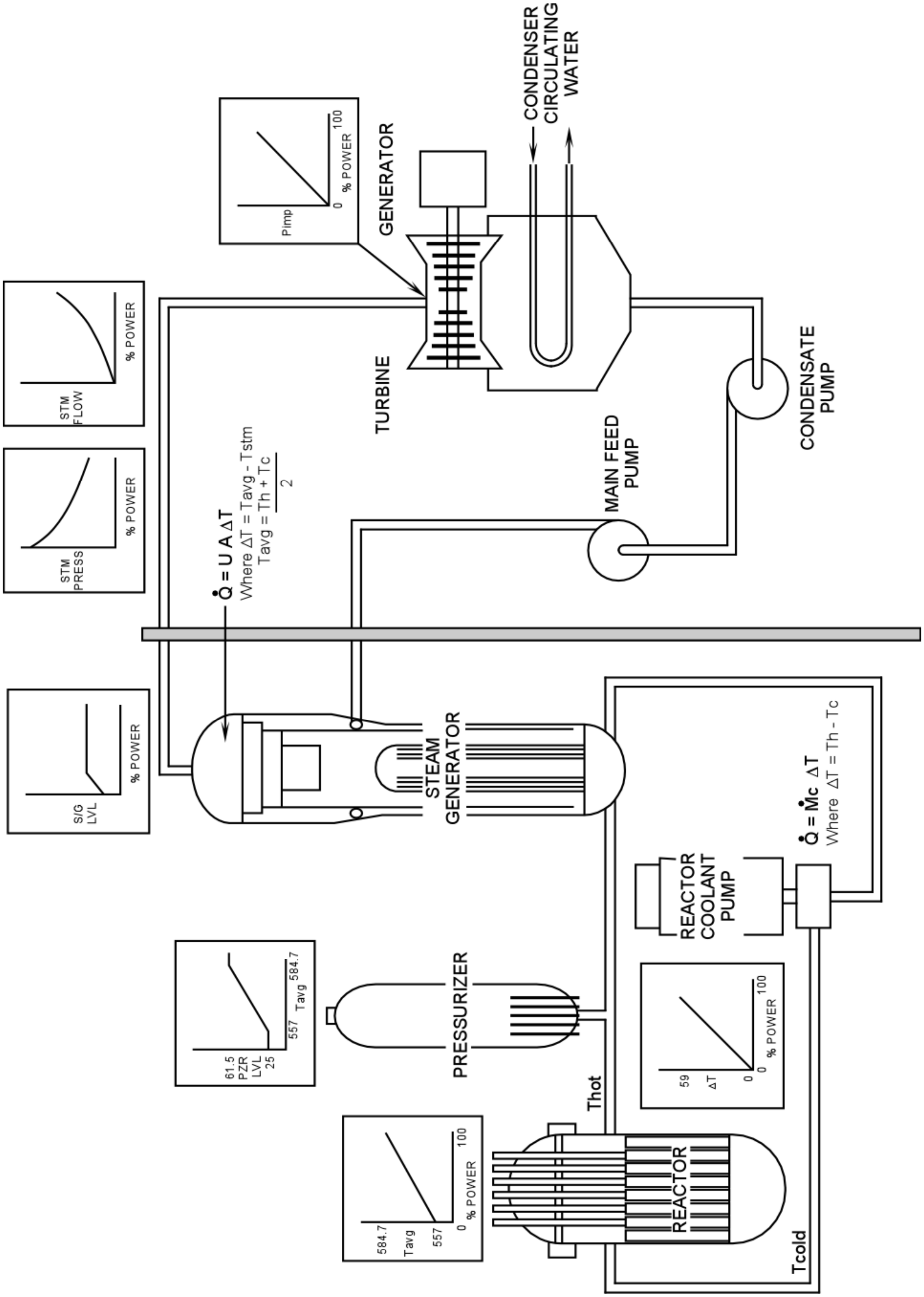


Figure 5-1 NSSS Response

This page intentionally blank

Transient 5.01 Ramp Load Increase, 50 - 100%, 5%/min

Initial Conditions:

BOL	Bank D Rod Position: 84 steps
T_{avg}: 570.3°F	Nuclear Power: 50%
PZR Pressure: 2235 psi	All control systems in automatic

Initiating Event: 5%/min load increase

Point Explanation

1. **Generator load** increases as the control valves open in response to the 5%/min load increase command input by the operator at the turbine EHC station.

2. **Bank D rod position** increases as outward motion is called for by the power mismatch (turbine load increasing relative to nuclear power) and temperature mismatch ($T_{ref} > T_{avg}$) circuits of the rod control system.

3. **Nuclear power** increases in response to the positive reactivity added by rod withdrawal.

4. **Pressurizer level** drops slightly at the start of the transient due to the slight cooling of the reactor coolant caused by the power mismatch, with turbine load > nuclear power. Note: The decrease in T_{avg} cannot be distinguished on the T_{avg} plot.

5. **T_{avg}** increases as the rods are withdrawn and nuclear power increases. Over the time interval in which T_{avg} is increasing, nuclear power exceeds turbine load.

6. **T_{ref}** increases with generator load (T_{ref} varies linearly with turbine P_{imp}).

7. **Charging flow** initially increases as the pressurizer level drops relative to the setpoint (which is unchanged at the start of the transient with no change in T_{avg}). Charging flow continues to increase with pressurizer level less than the level setpoint (both the level and the setpoint are constantly changing during the transient). Later, charging flow decreases when the actual level exceeds the setpoint.

8. **Bank D rod position** increases at a slower rate (and sometimes stops) during the 10 - 13-min interval as the total error input to the rod control system becomes small. During the intervals of constant rod position, the power mismatch circuit calls for rod insertion (due to the nuclear power increasing faster than the rate of turbine load increase), while the temperature mismatch circuit still calls for rod withdrawal ($T_{ref} > T_{avg}$). (Note: Although it is difficult to see on the plots, close inspection of the recorded data for this transient reveals that there are half-minute intervals in this time frame when the temperature error exceeds 2°F [enough for slow rod motion], but rods don't move. This means that during those intervals the power mismatch is of comparable magnitude and opposite sign.)

Point Explanation

9. **Steam flow** increases as the control valves open to increase load.

10. **Steam generator level** initially swells with the load increase.

11. **Feed flow** increases steadily throughout the transient as the steam generator water level control system attempts to match steam and feed flows. The slight decrease in feed flow very early in the transient is the result of the level control system response to the initial swell in steam generator level.

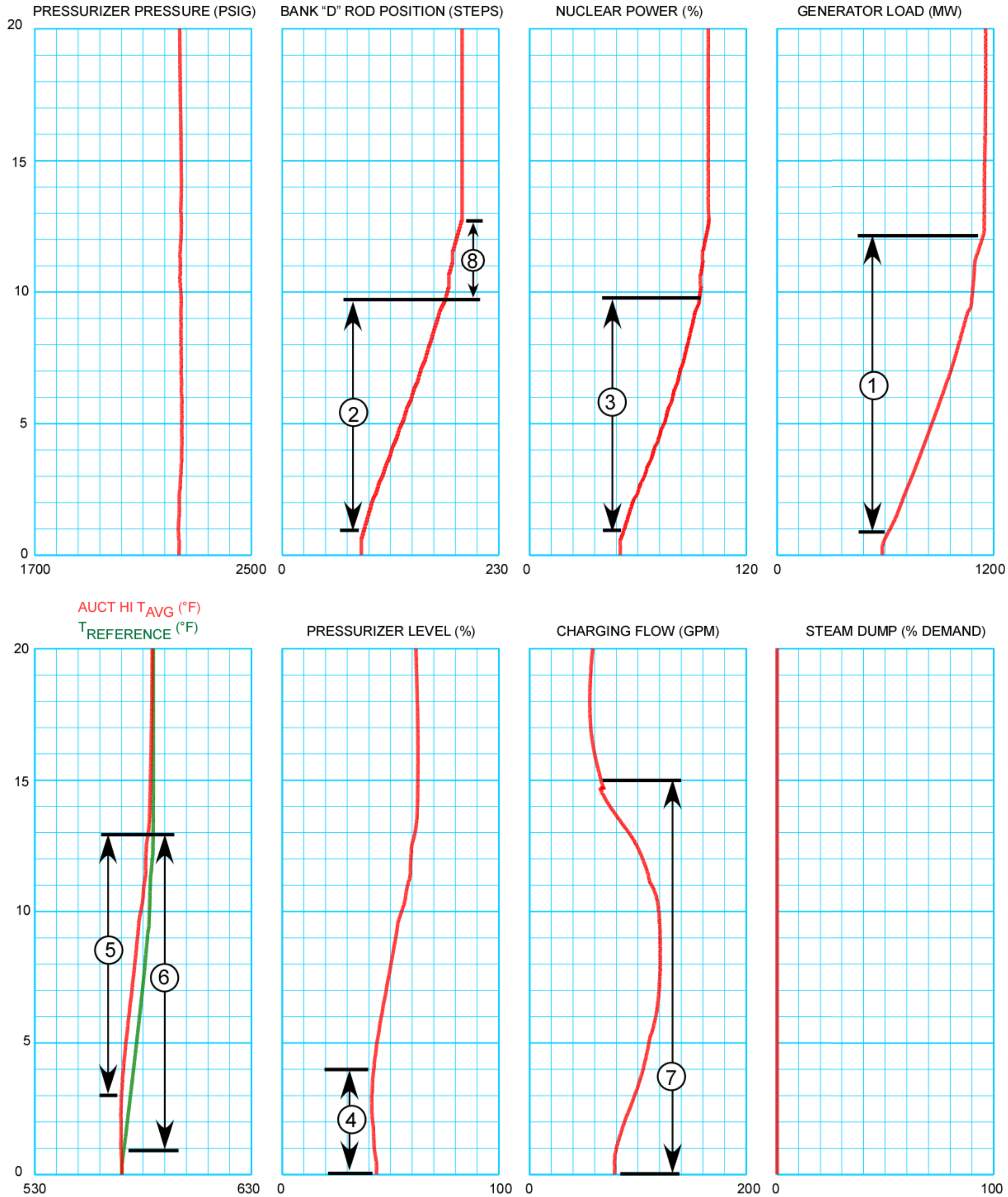
12. **Steam pressure** decreases as the control valves are opened. $\dot{Q} = UA(T_{avg} - T_{stm})$; T_{avg} is increasing and T_{stm} (also P_{stm}) is decreasing throughout the load change as $\dot{Q}$ increases.

Note 1: The starting bank D rod position of 84 steps is not typical for a plant operating at 50% power. The rods were initially diluted in to their starting positions so that the power change could be completed with all control systems in automatic and with T_{avg} maintained on program throughout. Trojan's relaxed axial offset technical specification is not very restrictive on AFD, so starting with the bank D rods ~ halfway in does not place the plant in an action statement.

Note 2: At steady state at the end of the transient, nuclear power = secondary load, so Tavg is unchanging. As for the reactivity balance, the negative reactivity added by the power defect associated with the power change is balanced by the positive reactivity added by the rod withdrawal, so that the net endpoint $\rho = 0$.

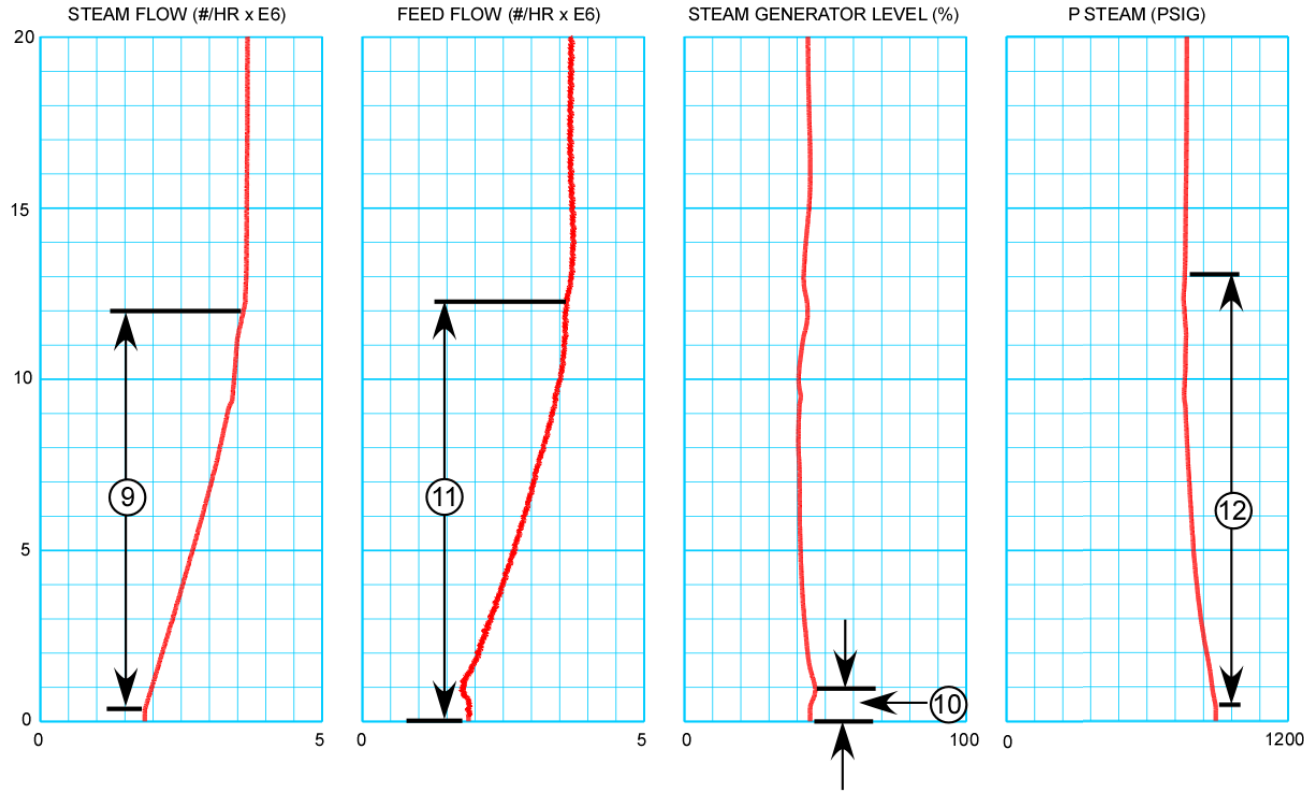
What this transient illustrates:

1. The plant response to a normal power increase controlled at the turbine EHC system station.
2. The actions of the rod control and pressurizer level control systems.
3. The initial swell in steam generator level associated with a power increase.
4. The programmed increase in T_{avg} and decrease in steam pressure associated with a power increase.



Transient 5.01 Ramp Load Increase, 50 - 100%, 5%/min

This page intentionally blank



TRANSIENT 5.01
RAMP LOAD INCREASE, 50-100%
@ 5%/MIN.

Initial Conditions
BOL
Tavg: 570.3°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 50%
Bank D Rod Position: 84 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
5%/min. load increase

Transient 5.01 Ramp Load Increase, 50 - 100%, 5%/min

This page intentionally blank

Transient 5.02 Ramp Load Decrease, 100 - 50%, 5%/min

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.3°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: 5%/min load decrease

Point **Explanation**

1. **Generator load** decreases as the control valves close in response to the 5%/min load decrease command input by the operator at the turbine EHC station. The "holdup" in the load decrease over the second and third minutes of the transient appears to be characteristic of the turbine EHC system.

2. **Bank D rod position** decreases as inward motion is called for by the power mismatch (turbine load decreasing relative to nuclear power) and temperature mismatch ($T_{ref} < T_{avg}$) circuits of the rod control system.

3. **Nuclear power** decreases in response to the negative reactivity added by rod insertion.

4. **Pressurizer level** increases slightly at the start of the transient due to the slight heatup of the reactor coolant caused by the power mismatch, with turbine load < nuclear power. Note: The increase in T_{avg} is difficult to distinguish on the T_{avg} plot.

5. $\mathbf{T_{avg}}$ decreases as the rods are inserted and nuclear power decreases. Over the time interval in which T_{avg} is decreasing, nuclear power is less than turbine load.

6. $\mathbf{T_{ref}}$ decreases with generator load (T_{ref} varies linearly with turbine P_{imp}).

7. **Charging flow** initially decreases as the pressurizer level rises relative to the setpoint (which is unchanged at the start of the transient with little change in T_{avg}). Charging flow later increases when the actual level decreases relative to the setpoint (both the level and the setpoint are constantly changing during the transient).

8. **Steam flow** decreases as the control valves close to decrease load.

9. **Feed flow** decreases as the steam generator water level control system closes the main feed regulating valves to match steam and feed flows.

10. **Steam pressure** increases as the control valves close. $\dot{Q} = UA(T_{avg} - T_{stm})$; T_{avg} is decreasing and T_{stm} (also P_{stm}) is increasing throughout the load change as $\dot{Q}$ decreases.

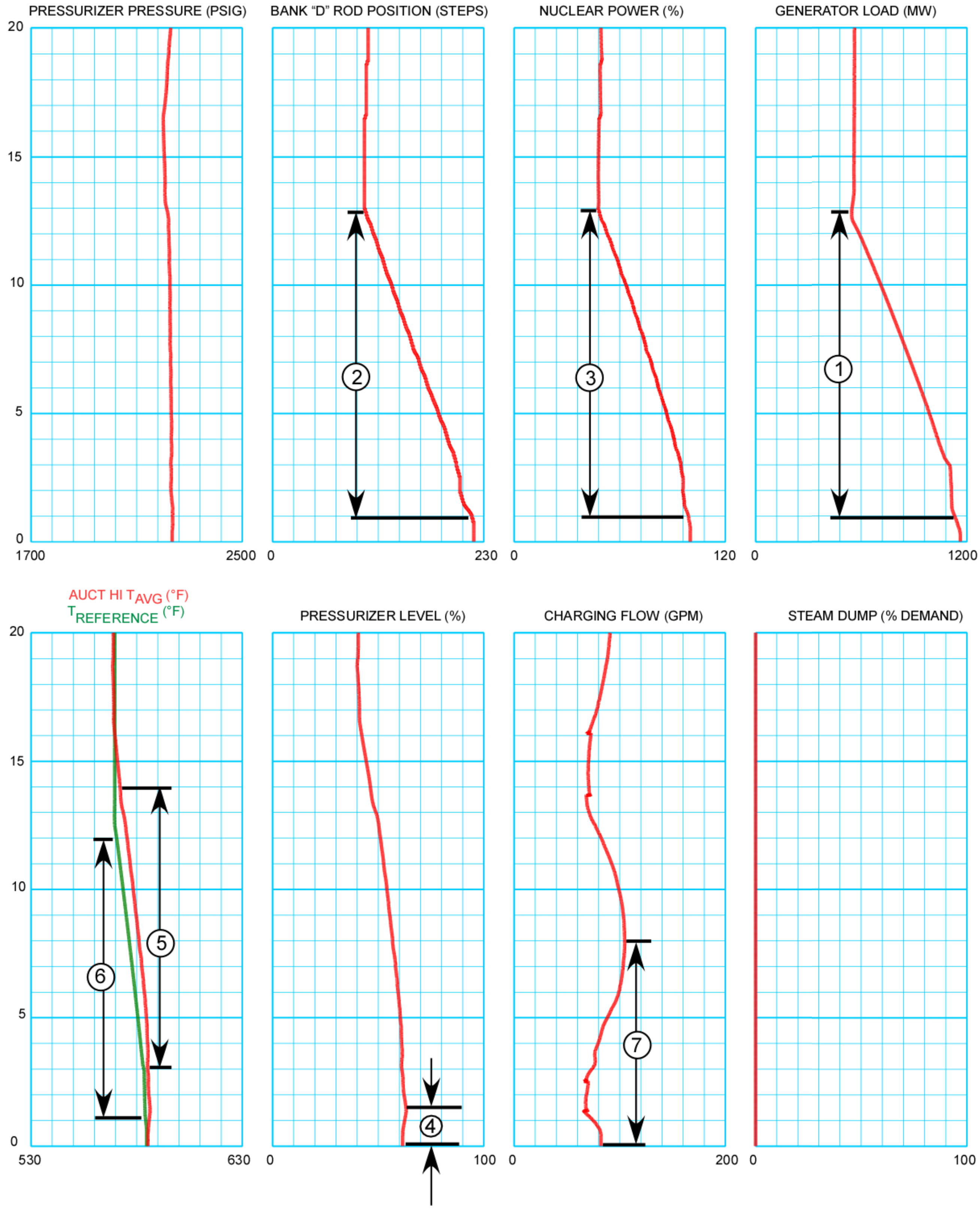
Transient 5.02 Ramp Load Decrease, 100 - 50%, 5%/min

Note 1: For this transient only, the turbine EHC system was modified to permit a rate-controlled load decrease. Typically, with a GE EHC system, a load decrease proceeds as fast as the control valves can close in response to the operator pushing the load decrease pushbutton or decreasing the valve position limit with the potentiometer.

Note 2: At steady state at the end of the transient, nuclear power = secondary load, so T_{avg} is more or less unchanging. As for the reactivity balance, the positive reactivity added by the power defect associated with the power change is balanced by the negative reactivity added by the rod insertion, so that the net endpoint $\rho = 0$.

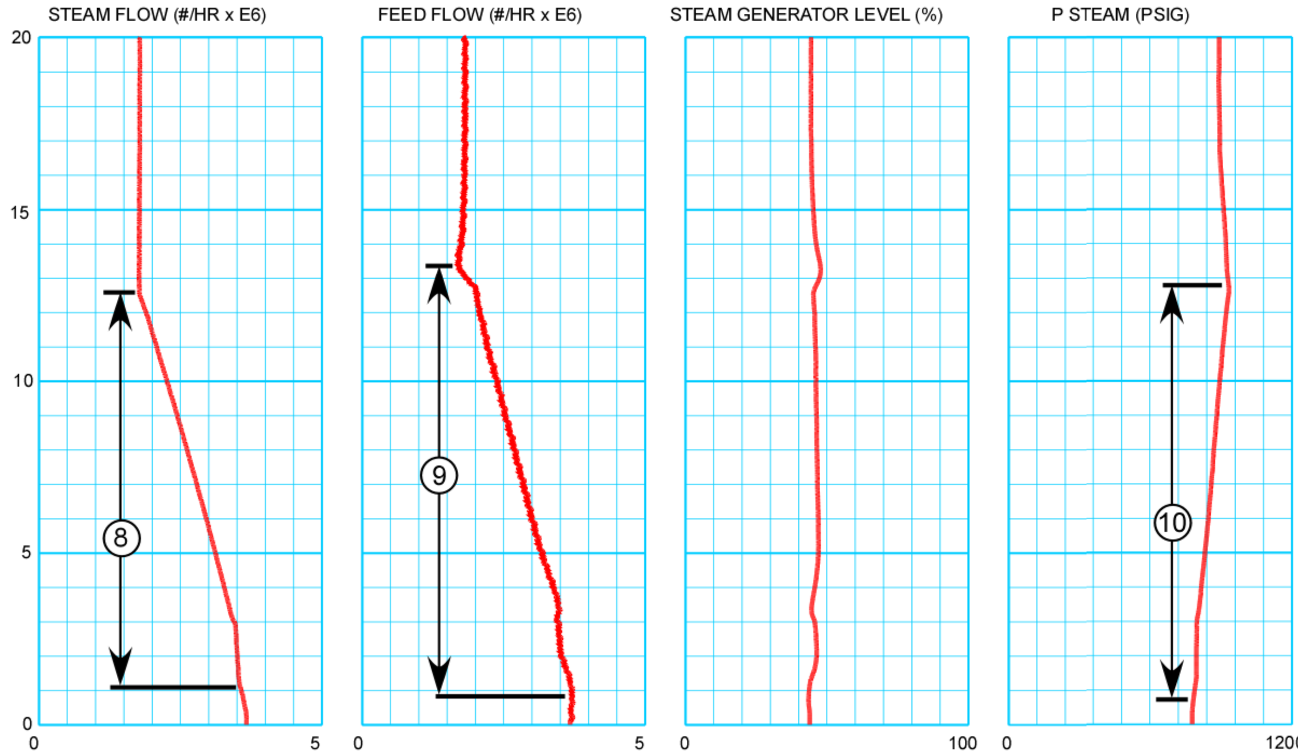
What this transient illustrates:

1. The plant response to a normal power decrease controlled at the turbine EHC system station.
2. The actions of the rod control and pressurizer level control systems.
3. The programmed decrease in T_{avg} and increase in steam pressure associated with a power decrease.



Transient 5.02 Ramp Load Decrease, 100 - 50%, 5%/min

This page intentionally blank



TRANSIENT 5.02
RAMP LOAD DECREASE, 100-50%
@ 5%/MIN.

Initial Conditions
BOL
Tavg: 585.3°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
5%/min. load decrease

Transient 5.02 Ramp Load Decrease, 100 - 50%, 5%/min

This page intentionally blank

Transient 5.03 Rapid Load Decrease, 100 - 90%

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Rapid closure of turbine control valves with the valve position limiter

Point Explanation

1. **Generator load** decreases as the control valves close in response to the rapid reduction of the valve position limiter potentiometer setpoint by the operator at the turbine EHC station. After the load change is complete, generator load responds to the subsequent changes in steam pressure because the load limit overrides the function of throttle pressure compensation.
2. $\mathbf{T_{ref}}$ decreases with generator load (T_{ref} varies linearly with turbine P_{imp}).
3. **Bank D rod position** decreases at or near the maximum rate (72 steps/min) as rapid inward rod motion is called for by the power mismatch (turbine load decreasing relative to nuclear power) and temperature mismatch ($T_{ref} < T_{avg}$) circuits of the rod control system.
4. **Nuclear power** decreases in response to the negative reactivity added by rod insertion and (to a small extent) by the increase in reactor coolant temperature (discussed in point 5 below).
5. **Pressurizer level** increases at the start of the transient due to the heatup of the reactor coolant caused by the power mismatch, with turbine load < nuclear power.
6. $\mathbf{T_{avg}}$ increases at the start of the transient due to the power mismatch, with turbine load < nuclear power. Note that, although it is somewhat difficult to see on the plots, the increase in the T_{avg} indication lags the increase in pressurizer level (which indicates the increase in actual T_{avg}) because of the RTD manifold arrangement and loop transport time.
7. **Charging flow** initially decreases as the pressurizer level rises relative to the level setpoint (which is unchanged at the start of the transient with T_{avg} > full-load T_{avg}).
8. **Steam flow** decreases as the control valves close to decrease load.
9. **Feed flow** decreases as the steam generator water level control system closes the main feed regulating valves to match steam and feed flows.
10. **Steam generator level** initially shrinks with the load decrease.

Point Explanation

11. **Feed flow** increases in response to the level error resulting from the initial shrink.

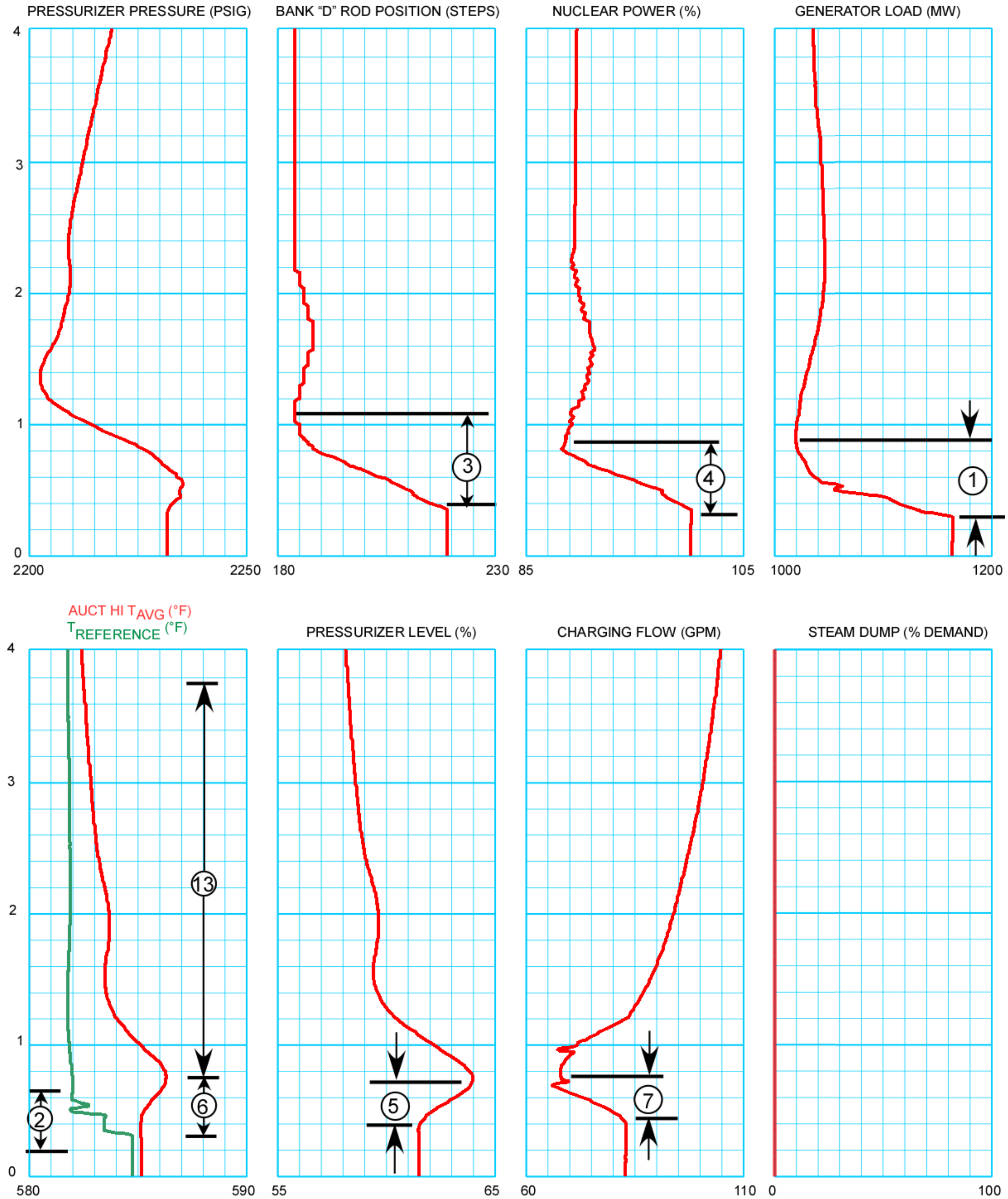
12. **Steam pressure** increases as the control valves close. $\dot{Q} = UA(T_{avg} - T_{stm})$; T_{stm} (also P_{stm}) is increasing as the turbine load (and $\dot{Q}$) is decreased. The increase in T_{stm} is exaggerated a bit by the increase in T_{avg}.

13. Rod insertion and a small mismatch between nuclear power and secondary load bring $\mathbf{T_{avg}}$ down toward T_{ref}.

Note: At steady state at the end of the transient, nuclear power = secondary load, so T_{avg} is more or less unchanging. As for the reactivity balance, the positive reactivity added by the power defect associated with the power change is balanced by the negative reactivity added by the rod insertion, so that the net endpoint $\rho = 0$.

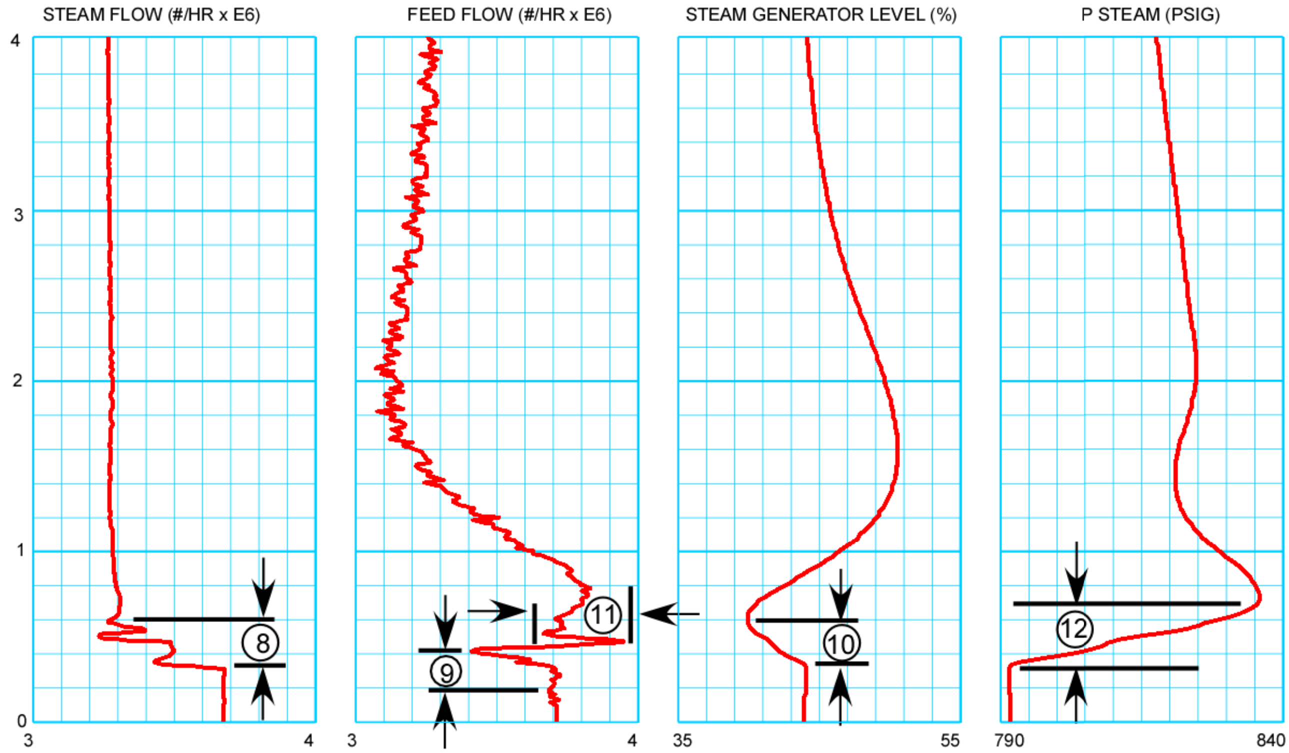
What this transient illustrates:

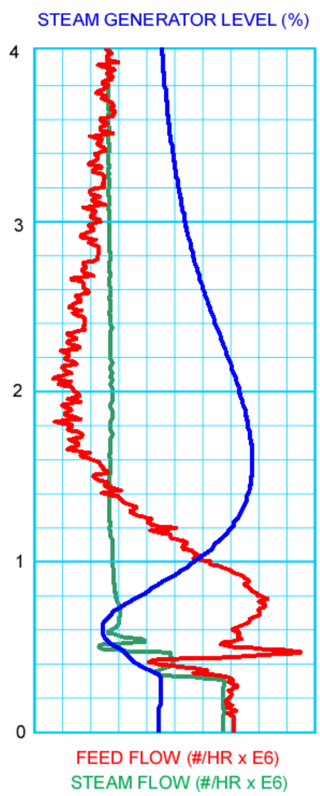
1. The plant response to a short, rapid power decrease initiated at the turbine EHC system station with the valve position limiter.
2. The actions of the rod control and pressurizer level control systems.
3. The initial shrink in steam generator water level associated with a power decrease, and the response of the steam generator water level control system.
4. The programmed decrease in T_{avg} and increase in steam pressure associated with a power decrease.



Transient 5.03 Rapid Load Decrease, 100 - 90%

This page intentionally blank





TRANSIENT 5.03
RAMP LOAD DECREASE, 100-90%

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Rapid closure of the turbine control valves with the valve position limiter

Transient 5.03 Rapid Load Decrease, 100 - 90%

This page intentionally blank

Transient 5.04 Rapid Load Decrease, 100 - 15%

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Rapid closure of turbine control valves with the valve position limiter

Point Explanation

1. **Generator load** rapidly decreases as the control valves close in response to the rapid reduction of the valve position limiter potentiometer setpoint by the operator at the turbine EHC station.

2. **T_{ref}** decreases with generator load (T_{ref} varies linearly with turbine P_{imp}).

3. **Bank D rod position** decreases at the maximum rate (72 steps/min) as rapid inward rod motion is called for by the power mismatch (turbine load decreasing rapidly relative to nuclear power) and temperature mismatch ($T_{ref} << T_{avg}$) circuits of the rod control system.

4. **Nuclear power** decreases rapidly in response to the negative reactivity added by rod insertion and by the increase in reactor coolant temperature, as discussed in point 7.

5. **Pressurizer level** increases at the start of the transient due to the heatup of the reactor coolant caused by the power mismatch, with secondary load < nuclear power.

6. **Pressurizer pressure** rises as the steam bubble is squeezed by the thermal expansion of the reactor coolant. The pressurizer PORVs lift twice to mitigate the pressure rise.

7. **T_{avg}** increases at the start of the transient due to the power mismatch, with secondary load < nuclear power. The increase in T_{avg} is limited by the actuation of the steam dumps, which limits the total primary-to-secondary power imbalance.

8. **Charging flow** initially decreases sharply as the pressurizer level rises relative to the level setpoint (which is unchanged at the start of the transient with T_{avg} > full-load T_{avg}).

9. **Steam dump demand** increases rapidly with the rapid reduction in T_{ref} (the T_{avg} - T_{ref} difference rapidly increases). All 12 steam dump valves open fully, as discussed in point 13.

Point Explanation

10. **Steam flow** initially decreases rapidly as the control valves close to decrease load.

11. **Feed flow** decreases as the steam generator water level control system closes the main feed regulating valves to match steam and feed flows.

12. **Steam generator level** initially shrinks with the rapid load decrease.

13. **Steam flow** increases (stops decreasing) as all steam dump valves open fully with a loss-of-load arming signal (from the rapid reduction in turbine load) and maximum demand.

14. **Feed flow** rapidly increases in response to the rapid increase in steam flow and to the level error resulting from the initial shrink.

15. **Steam pressure** increases as the control valves close. $\dot{Q} = UA(T_{avg} - T_{stm})$; T_{stm} (also P_{stm}) is increasing as the turbine load (and $\dot{Q}$) is decreased. Note that the rise in steam pressure is limited by steam dump operation; after the dumps actuate, total secondary load is always > turbine load.

16. **Bank D rod position** stays constant for this period of time. At this point, the initial large input to the power mismatch circuit from the reduction in turbine load has died off, nuclear power has been decreasing faster than turbine load, and T_{avg} is > T_{ref}. The total error input to the rod control system is < 1°F, and rod motion stops.

17. **Bank D rod position** decreases again, at a slower rate than earlier in the transient, with $T_{avg} > T_{ref}$. This negative reactivity causes reactor power to be lower than secondary power, which reduces T_{avg}. When T_{avg} drops to within 5°F of T_{ref}, the steam dumps are fully closed. Control rod motion has the overall effect of closing the steam dump valves.

Note 1: As T_{avg} decreases, the steam dump demand trends toward 0, and secondary load will return to turbine load only. The steam dumps have been armed by a loss-of-load arming signal and will remain armed until they are reset by the operator.

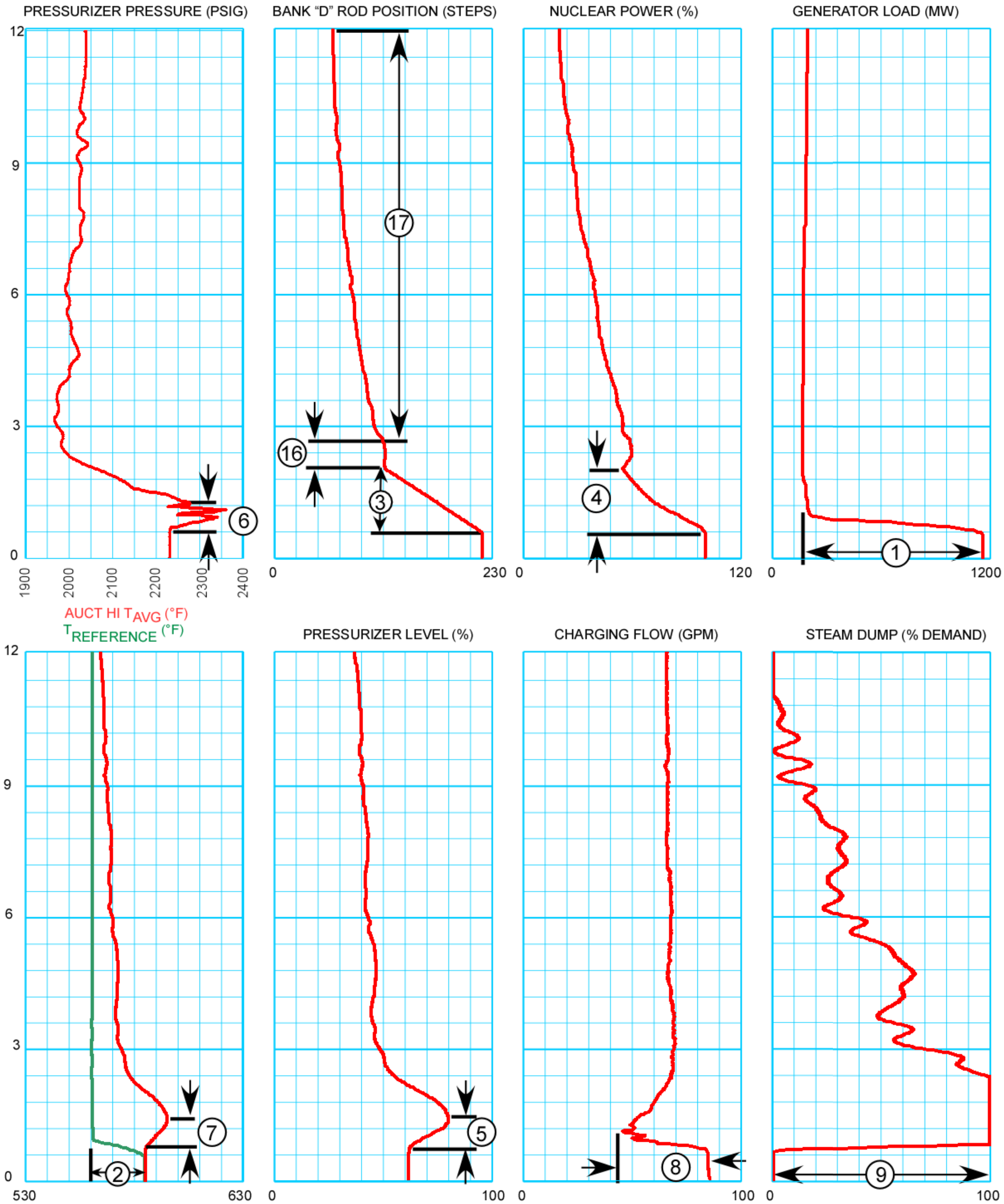
Note 2: This transient is beyond the design of the system. Automatic control rod motion is designed to accommodate a 10% step change and the steam dumps have a 40% capacity. Any load rejection > 50% may result in a reactor trip, most likely on high pressurizer pressure or low steam generator level. The trip was avoided by limiting the rate of secondary power reduction.

Transient 5.04 Rapid Load Decrease, 100 - 15% (cont'd)

What this transient illustrates:

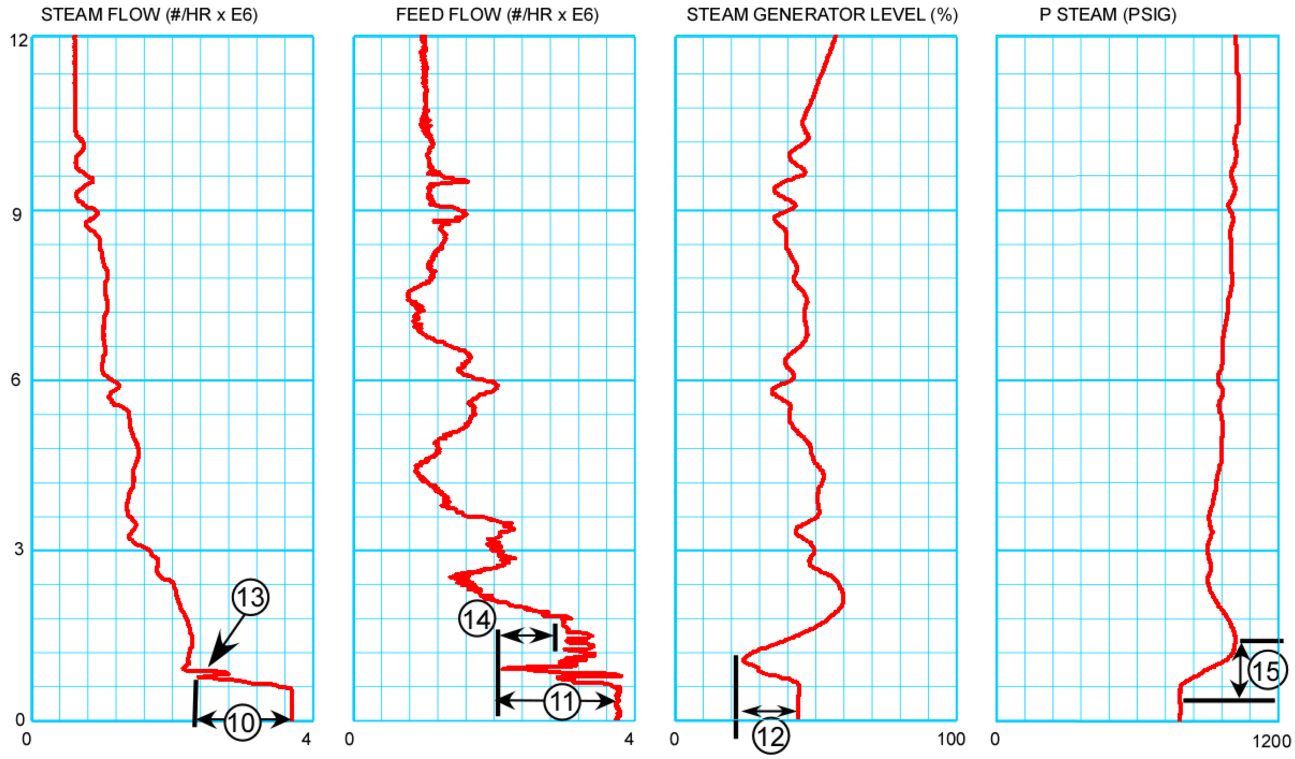
1. The plant response to a rapid power decrease initiated at the turbine EHC system station with the valve position limiter.
2. The actions of the rod control ,pressurizer level control, and pressurizer pressure control systems.
3. The initial shrink in steam generator water level associated with a power decrease, and the response of the steam generator water level control system.
4. The actions of the steam dump control system in response to a large loss of load.
5. The increase in T_{avg} when nuclear power >> secondary load.
6. The programmed decrease in T_{avg} and increase in steam pressure associated with a power decrease.
7. Control rod motion (or some negative reactivity) is necessary to cause the steam dump valves to close after a load rejection causes them to open. The steam dumps only act to limit the rise in T_{avg}.

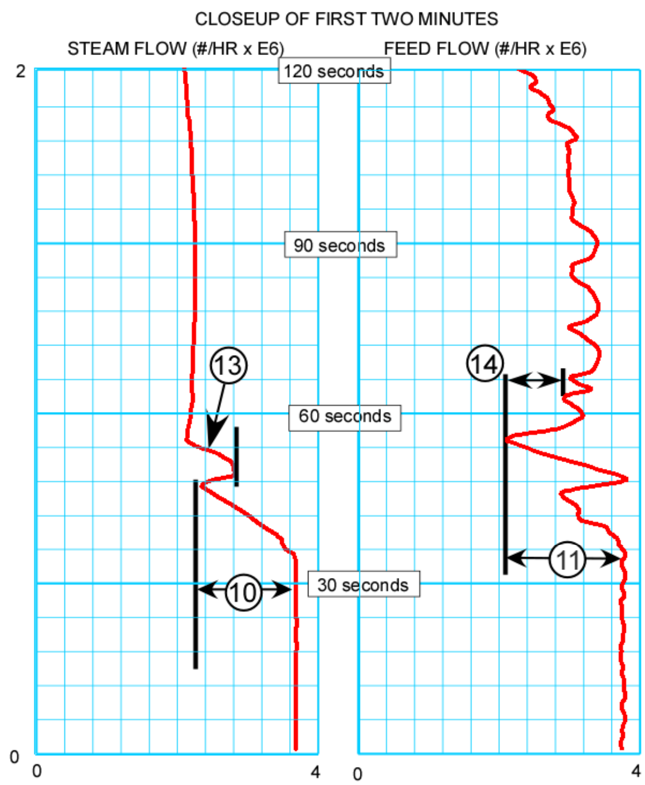
This page intentionally blank



Transient 5.04 Rapid Load Decrease, 100 - 15%

This page intentionally blank





TRANSIENT 5.04
RAPID LOAD DECREASE, 100-15%

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Rapid closure of the turbine control valves with the valve position limiter

Transient 5.04 Rapid Load Decrease, 100 - 15%

This page intentionally blank

Transient 5.11 Manual Reactor Trip

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Operator depresses manual trip pushbutton

<u>**Point**</u> <u>**Explanation**</u>

1. **Bank D rod position** falls to 0 in ~ 2 sec when the opening of the reactor trip breakers kills power to all rod drive system coils.

2. **Nuclear power** initially decreases rapidly in concert with the prompt drop in neutron flux associated with the trip. At ~ 15 sec, the rate of power decrease follows the decay of delayed neutron precursors.

3. **Generator load** decreases rapidly to 0 when the turbine trips. The turbine trip input comes from the P-4 contact of the reactor protection system.

4. **Pressurizer level** decreases with the increase in reactor coolant density. Reactor coolant temperature is decreasing rapidly in response to the mismatch between nuclear power (decreasing to decay heat level with the trip) and secondary load (momentarily at maximum steam dump capacity). The minimum level reached is 18.9%.

5. The decrease in **pressurizer pressure** reflects the expansion of the steam bubble associated with the drop in pressurizer level.

6. **Charging flow** responds to pressurizer level control system commands. Immediately following the trip, charging flow initially increases with pressurizer level < the level setpoint, then sharply decreases with level > the level setpoint (pressurizer level is decreasing with the increase in reactor coolant density, and the level setpoint is decreasing with the decrease in auctioneered high T_{avg}, but level and level setpoint are not in step with each other). Later, the large "bulge" in charging flow reflects the decrease in pressurizer level to below the no-load level setpoint (25%).

7. **Steam dump demand** follows the T_{avg} - 557°F (no-load T_{ref}) difference. The turbine trip causes the output of the turbine-trip controller to be selected as steam dump demand. The steam dump demand increases to maximum immediately after the trip with the large T_{avg} – 557°F difference, then decreases as the steam dump actuation reduces T_{avg} below the no-load T_{ref} value.

Point Explanation

8. **Steam flow** (1) decreases to 0 as the turbine steam admission valves close with the turbine trip, then (2) increases as the steam dump valves blast open in response to the turbine trip arming signal and large demand, and finally (3) returns to 0 as the steam dump valves modulate closed in response to the reduction in demand.

9. **Steam generator level** shrinks to below the bottom of the narrow-range indicating range with the closure of the turbine steam admission valves.

10. The steam generator level control system at first decreases **feed flow** in response to the reduction in steam flow with the trip, and then increases feed flow in response to (1) the increase in steam flow with steam dump actuation and (2) the steam generator level error (level < setpoint) resulting from the trip-induced shrink.

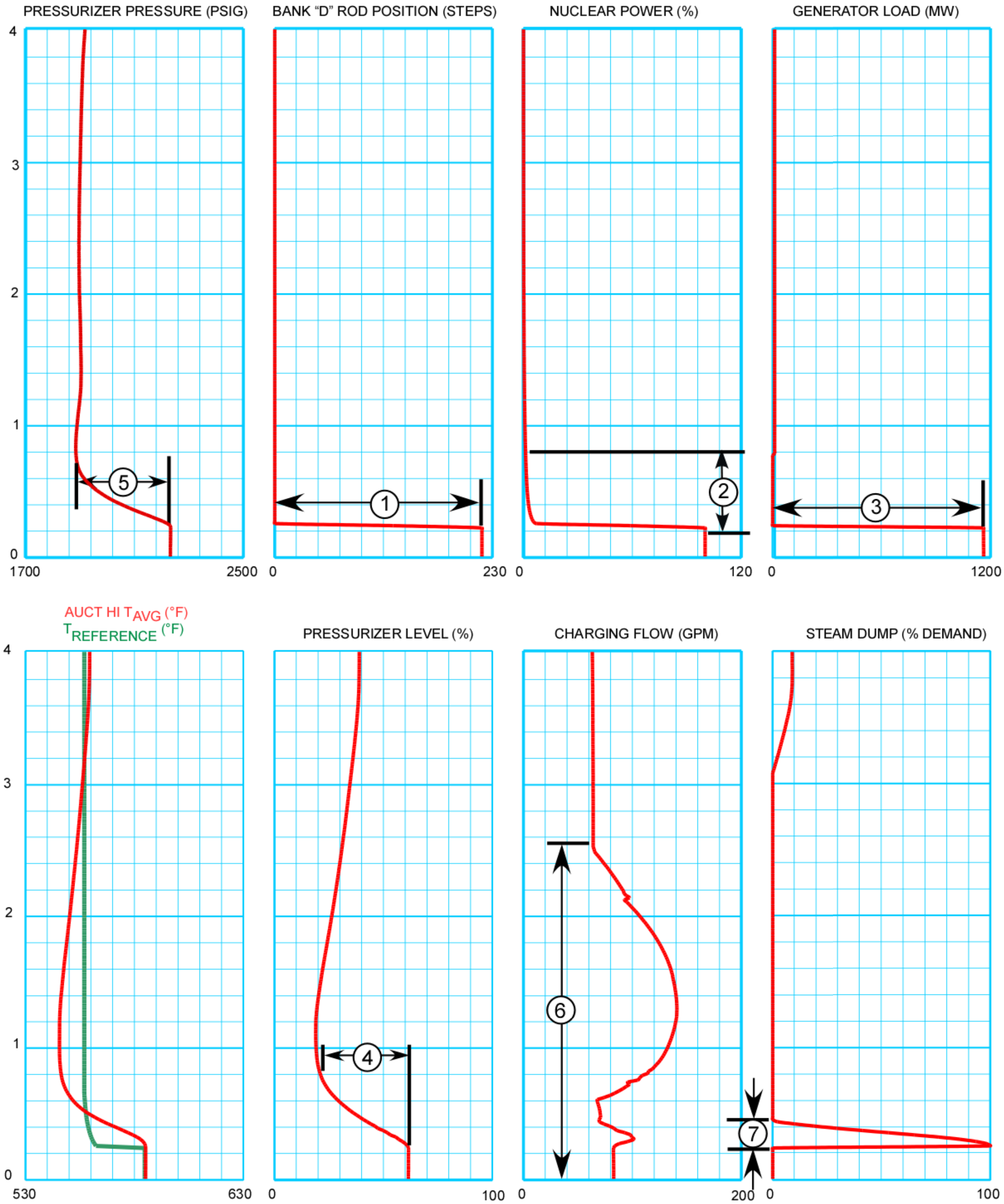
11. The rapid reduction in **feed flow** reflects feedwater isolation on reactor trip coincident with low T_{avg} (564°F). All feedwater regulating valves and feedwater isolation valves close with the isolation signal.

12. The indicated **feed flow** reflects operation of the AFW system.

13. **Steam pressure** tracks toward P_{sat} (1092 psig) for no-load T_{avg}. Decay heat and reactor coolant pump heat increase the reactor coolant temperature and, in turn, steam pressure. Toward the end of the interval, the reactor coolant temperature has increased to the no-load T_{avg} value, and the steam dump valves are modulating to maintain the no-load T_{avg}/steam pressure. Ultimately, the rate of energy removal from the reactor coolant by the SGs/steam dumps equals the rate of energy addition from decay heat and the reactor coolant pumps.

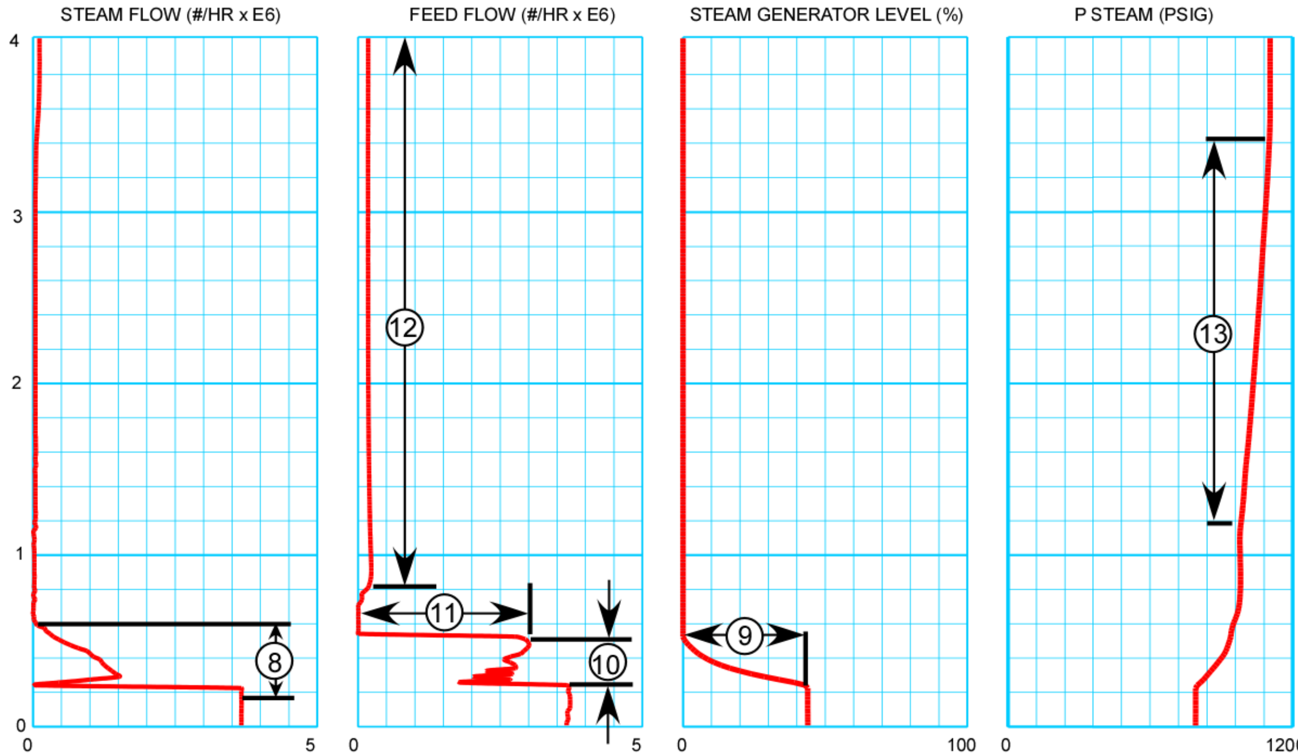
What this transient illustrates:

1. The plant response to a reactor trip.
2. The actions of the pressurizer level control system.
3. The initial shrink in steam generator water level associated with a turbine trip, and the response of the steam generator water level control system.
4. The actions of the steam dump control system in response to a turbine trip and the maintenance of no-load T_{avg}/steam pressure after a turbine trip.



Transient 5.11 Manual Reactor Trip

This page intentionally blank



TRANSIENT 5.11
MANUAL REACTOR TRIP

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Operator depresses the manual reactor trip pushbutton

Transient 5.11 Manual Reactor Trip

This page intentionally blank

Transient 5.12 Rapid Load Decrease, 100 - 50%, Rods In Manual

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	Rod bank selector switch placed in manual

Initiating Event: Rapid closure of turbine control valves with the valve position limiter

<u>**Point**</u> <u>**Explanation**</u>

1. **Generator load** rapidly decreases as the control valves close in response to the rapid reduction of the valve position limiter potentiometer setpoint by the operator at the turbine EHC station.

2. $\mathbf{T_{ref}}$ decreases with generator load (T_{ref} varies linearly with turbine P_{imp}).

3. **Bank D rod position** does not change with the rod control system in manual.

4. **Pressurizer level** increases at the start of the transient due to the heatup of the reactor coolant caused by the power mismatch, with secondary load < nuclear power.

5. **Pressurizer pressure** rises as the steam bubble is squeezed by the thermal expansion of the reactor coolant. PORV operation mitigates the pressure rise.

6. $\mathbf{T_{avg}}$ increases at the start of the transient due to the power mismatch, with secondary load < nuclear power. Note that, although it is somewhat difficult to see on the plots, the increase in the T_{avg} indication lags the increase in pressurizer level (which indicates the increase in actual T_{avg}) because of the RTD manifold arrangement and loop transport time. The increase in T_{avg} is limited by the actuation of the steam dumps, which limits the total primary-to-secondary power imbalance.

7. **Nuclear power** decreases in response to the negative reactivity added by the increase in T_{avg}. Nuclear power drops to about 93% and stays approximately constant over the last three minutes of the transient, when there is little change in reactivity effects.

8. **Charging flow** initially decreases sharply as the pressurizer level rises relative to the level setpoint (which is unchanged at the start of the transient with T_{avg} > full-load T_{avg}).

9. **Steam dump demand** increases rapidly with the rapid reduction in T_{ref} (the T_{avg} - T_{ref} difference rapidly increases). The maximum steam dump demand (T_{avg} - T_{ref} = 16.4°F) is reached at 23 sec. All 12 steam dump valves open fully, as discussed in point 13.

Transient 5.12 Rapid Load Decrease, 100 - 50%, Rods In Manual (cont'd)

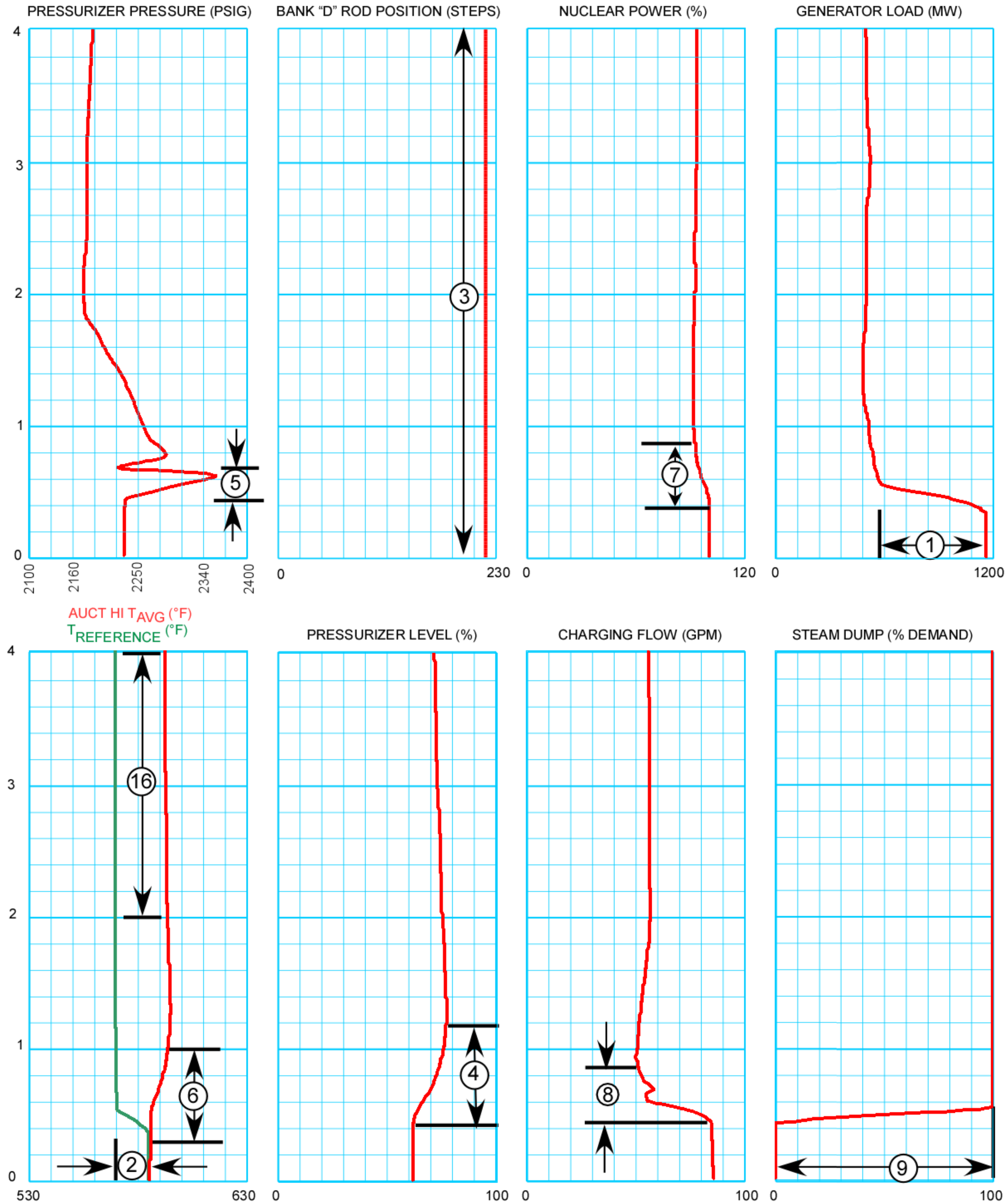
Point **Explanation**

10. **Steam flow** initially decreases rapidly as the control valves close to decrease load.
11. **Feed flow** decreases as the steam generator water level control system closes the main feed regulating valves to match steam and feed flows.
12. **Steam generator level** initially shrinks with the rapid load decrease.
13. **Steam flow** rapidly increases as all steam dump valves open fully with a loss-of-load arming signal (from the rapid reduction in turbine load) and maximum demand.
14. **Feed flow** rapidly increases in response to the rapid increase in steam flow and to the level error resulting from the initial shrink.
15. **Steam generator level** increases with the increased feed flow in response to the low level after the initial shrink (note that feed flow significantly exceeds the steam flow value of ~ 3 X 10^6 lbm/hr for the last half of the first minute). The large overshoot in level is a result of the regulating valve's PI controller response to the interval when level was much less than setpoint.
16. $\mathbf{T_{avg}}$ (and $\mathbf{T_{avg} - T_{ref}}$) stays constant over the last minutes of the transient, as nuclear power is equal to total secondary load. Turbine load = ~ 40%; nuclear power = ~ 90%.

Note: At steady state at the end of the transient, nuclear power = total secondary load (turbine load + steam dumps), so T_{avg} is unchanging. As for the reactivity balance, the positive reactivity added by the decrease in fuel temperature associated with the relatively small power change is balanced by the negative reactivity added by the increase in coolant temperature, so that the net endpoint $\rho = 0$.

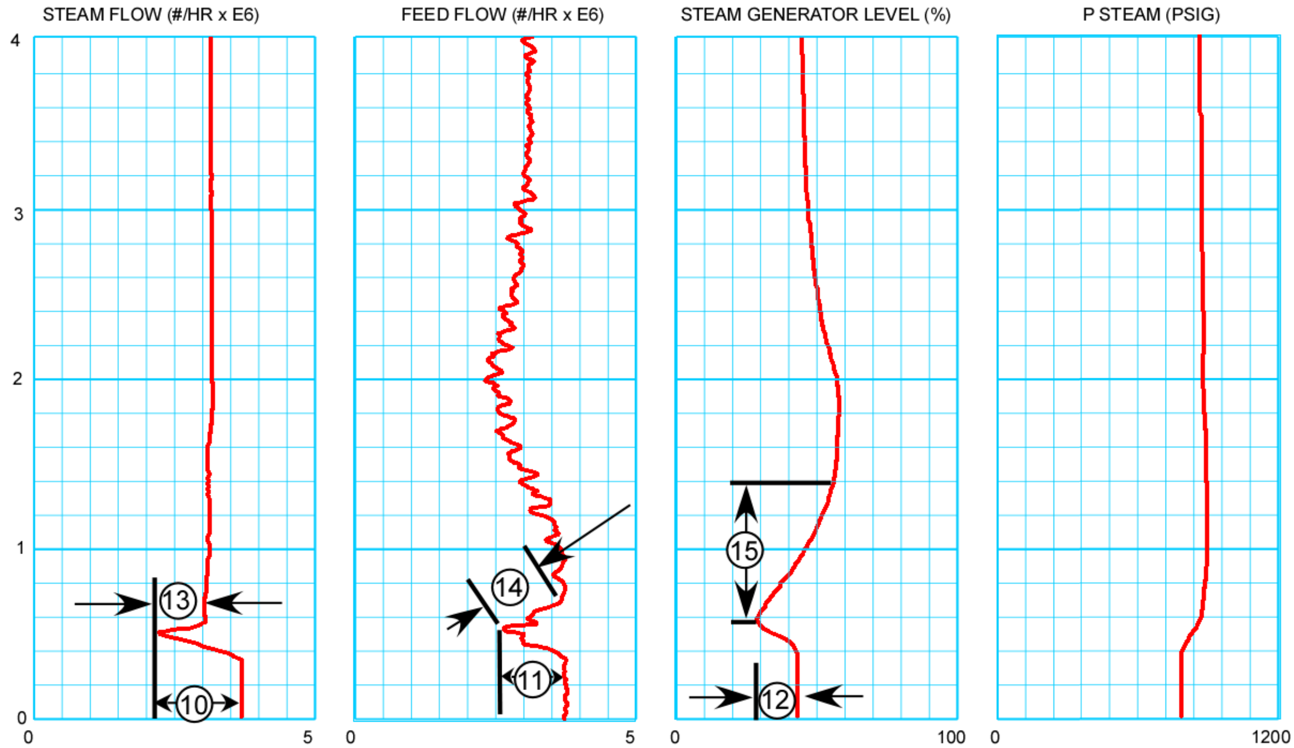
What this transient illustrates:

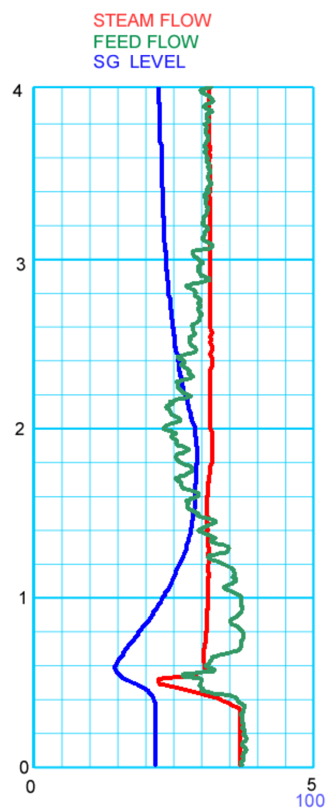
1. The generator load response to a rapid power decrease initiated at the turbine EHC system station with the valve position limiter.
2. The actions of the pressurizer level control and pressurizer pressure control systems.
3. The initial shrink in steam generator water level associated with a power decrease, and the ensuing recovery in SG level as the boiling rate increases with increasing T_{avg}.
4. The actions of the steam dump control system in response to a large loss of load, and steam dumps serving as a significant portion of total secondary load. The steam dumps do not restore T_{avg} to program value, they only limit the rise in T_{avg}. Adding negative reactivity will restore T_{avg} and close the steam dumps.
5. The increase in T_{avg} when nuclear power >> secondary load.



Transient 5.12 Rapid Load Decrease, 100 - 50%, Rods In Manual

This page intentionally blank





TRANSIENT 5.12
RAPID LOAD DECREASE, 100-50%
RODS IN MANUAL

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
Rod bank selector switch placed in manual

Initiating Event:
Rapid closure of the turbine control valves with the valve position limiter

Transient 5.12 Rapid Load Decrease, 100 - 50%, Rods In Manual

This page intentionally blank

Transient 5.13 Rapid Load Decrease, 100 - 50%, Steam Dumps Off

Initial Conditions:

BOL
T_{avg}: 585.1°F
PZR Pressure: 2235 psig
Bank D Rod Position: 219 steps
Nuclear Power: 100%
Steam dump bypass interlock switches placed in off position

Initiating Event: Rapid closure of turbine control valves with the valve position limiter

Point Explanation

1. **Generator load** rapidly decreases as the control valves close in response to the rapid reduction of the valve position limiter potentiometer setpoint by the operator at the turbine EHC station.

2. **Bank D rod position** decreases at the maximum rate (72 steps/min) as rapid inward rod motion is called for by the power mismatch (turbine load decreasing rapidly relative to nuclear power) and temperature mismatch ($T_{ref} << T_{avg}$) circuits of the rod control system.

3. **Pressurizer level** increases greatly at the start of the transient due to the heatup of the reactor coolant caused by the power mismatch, with turbine load < nuclear power.

4. **Pressurizer pressure** rises as the steam bubble is squeezed by the thermal expansion of the reactor coolant. The pressure increase is initially turned by PORV actuation. The second rise in pressure is too severe to be mitigated by PORV operation, so the reactor trips on high pressurizer pressure (2385 psig).

5. **T_{avg}** increases at the start of the transient due to the power mismatch, with turbine load < nuclear power. Rod insertion is not as effective as steam dump actuation in limiting the increase in T_{avg}, since this heatup results in a reactor trip, but transient 5.12 (Rapid Load Decrease, 100 - 50%, Rods in Manual) did not result in a reactor trip.

6. **Nuclear power** decreases in response to the negative reactivity added by the rod insertion and the increase in T_{avg}.

7. **Steam dump demand** increases rapidly to maximum with the rapid reduction in T_{ref} (the T_{avg} - T_{ref} difference rapidly increases). No steam dump valves open, as shown by the steam flow trace, as the steam dump bypass interlock switches have been placed in the off positions.

8. **Steam flow** initially decreases rapidly as the control valves close to decrease load. Steam flow rises slightly after the load reduction due to the rise in steam pressure with constant turbine valve position.

9. **Steam generator level** initially shrinks with the rapid load decrease.

Point Explanation

10. **Steam pressure** increases rapidly with the closure of the turbine control valves and the large increase in T_{avg}. Once the new control valve position is reached, turbine load and heat transfer in the SGs are essentially constant, so steam pressure must constantly increase. $\dot{Q} = UA(T_{avg} - T_{stm})$; with an increasing T_{avg}, T_{stm} (and P_{stm}) must increase for a constant $\dot{Q}$.

11. **Steam generator level** increases with the increased feed flow in response to the low level after the initial shrink.

12. The reactor trips (as indicated by the step drop in **bank D rod position**) with pressurizer pressure greater than the setpoint (2385 psig).

13. **Feed flow** drops as main feedwater is isolated on low T_{avg} with a reactor trip. The unavailability of steam dumps causes the post-trip feedwater isolation to be delayed.

14. **Feed flow** reflects AFW system operation after the feedwater isolation.

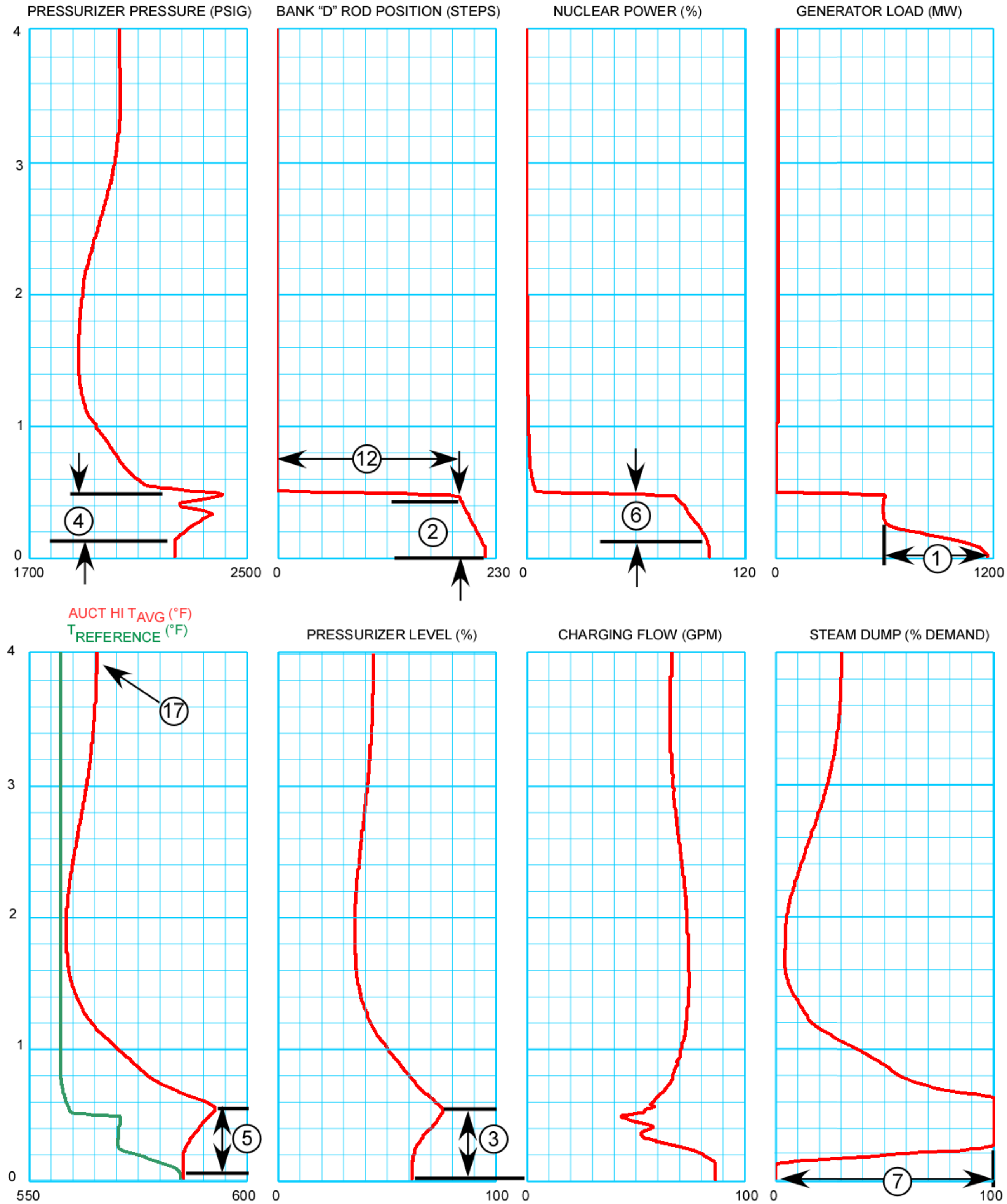
15. **Steam pressure** increases above the setpoint for the lowest set safety valve (1170 psig). This is an expected response to a reactor trip from high power with no steam dump operation.

16. The large increase in **steam flow** after the trip reflect opening of the PORV and the lowest set safety valve, with most of the flow attributable to the MSSV.

17. With the steam dumps unavailable, $\mathbf{T_{avg}}$ will ultimately be maintained at the saturation temperature for the PORV lift pressure instead of at no-load T_{avg}.

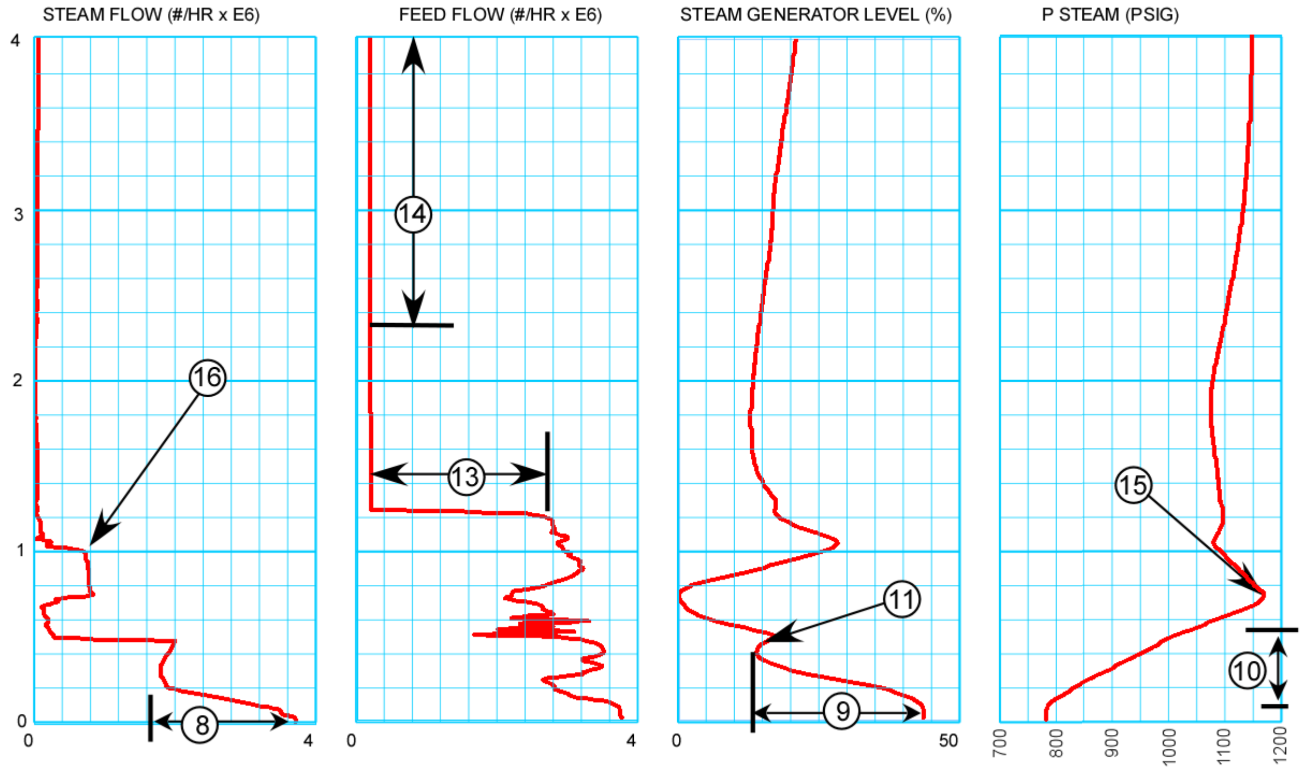
What this transient illustrates:

1. The generator load response to a rapid power decrease initiated at the turbine EHC system station with the valve position limiter.
2. The actions of the rod control, pressurizer level control, and pressurizer pressure control systems.
3. The initial shrink in steam generator water level associated with a power decrease.
4. The increase in T_{avg} when nuclear power >> secondary load.
5. That rod insertion (starting with bank D rods almost completely withdrawn) is not as effective as steam dump actuation in limiting the increase in T_{avg} with nuclear power >> secondary load.



Transient 5.13 Rapid Load Decrease, 100 - 50%, Steam Dumps Off

This page intentionally blank



TRANSIENT 5.13
RAPID LOAD DECREASE, 100-50%
STEAM DUMPS OFF

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
Steam dumps turned off

Initiating Event:
Rapid closure of the turbine control valves with the valve position limiter

Transient 5.13 Rapid Load Decrease, 100 - 50%, Steam Dumps Off

This page intentionally blank

Transient 5.14 Rapid Load Decrease, 100 - 50%, Steam Dumps Off, Rods In Manual

Initial Conditions:

BOL
T_{avg}: 585.1°F
PZR Pressure: 2235 psig
Bank D Rod Position: 219 steps
Nuclear Power: 100%
Steam dump bypass interlock switches placed in off position
Rod bank selector switch placed in manual

Initiating Event: Rapid closure of turbine control valves with the valve position limiter

Point Explanation

1. **Generator load** rapidly decreases as the control valves close in response to the rapid reduction of the valve position limiter potentiometer setpoint by the operator at the turbine EHC station.

2. **Pressurizer level** increases greatly at the start of the transient due to the heatup of the reactor coolant caused by the power mismatch, with secondary load < nuclear power.

3. **Pressurizer pressure** rises as the steam bubble is squeezed by the thermal expansion of the reactor coolant. The pressure increase is initially turned by PORV actuation. The second rise in pressure is too severe to be mitigated by PORV operation, so the reactor trips on high pressurizer pressure (2385 psig).

4. **T_{avg}** increases at the start of the transient due to the power mismatch, with secondary load < nuclear power.

5. **Nuclear power** decreases in response to the negative reactivity added by the increase in T_{avg}.

6. **Steam flow** initially decreases rapidly as the control valves close to decrease load.

7. **Steam generator level** initially shrinks with the rapid load decrease.

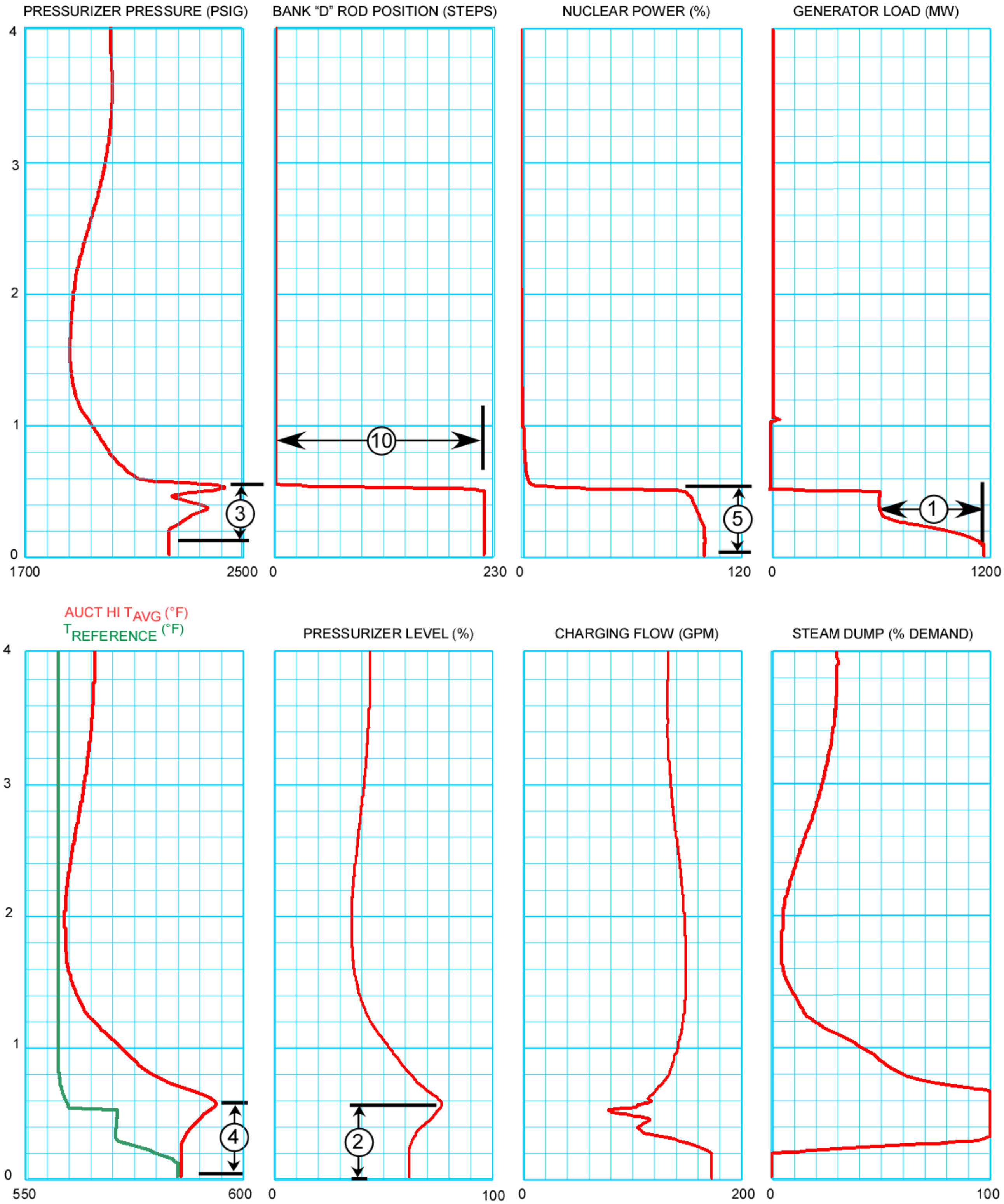
8. **Steam pressure** increases rapidly with the closure of the turbine control valves and the large increase in T_{avg}. Once the new control valve position is reached, turbine load and heat transfer in the SGs are essentially constant, so steam pressure must constantly increase. $\dot{Q} = UA(T_{avg} - T_{stm})$; with an increasing T_{avg}, T_{stm} (and P_{stm}) must increase for a constant $\dot{Q}$.

<u>Point</u> **<u>Explanation</u>**

9. **Steam generator level** increases with the increased feed flow in response to the low level after the initial shrink.
10. The reactor trips (as indicated by the step drop in **bank D rod position**) on high pressurizer pressure.
11. The large increase in **steam flow** after the trip reflects openings of the PORV and the lowest set safety valves, with most of the flow attributable to the MSSVs.

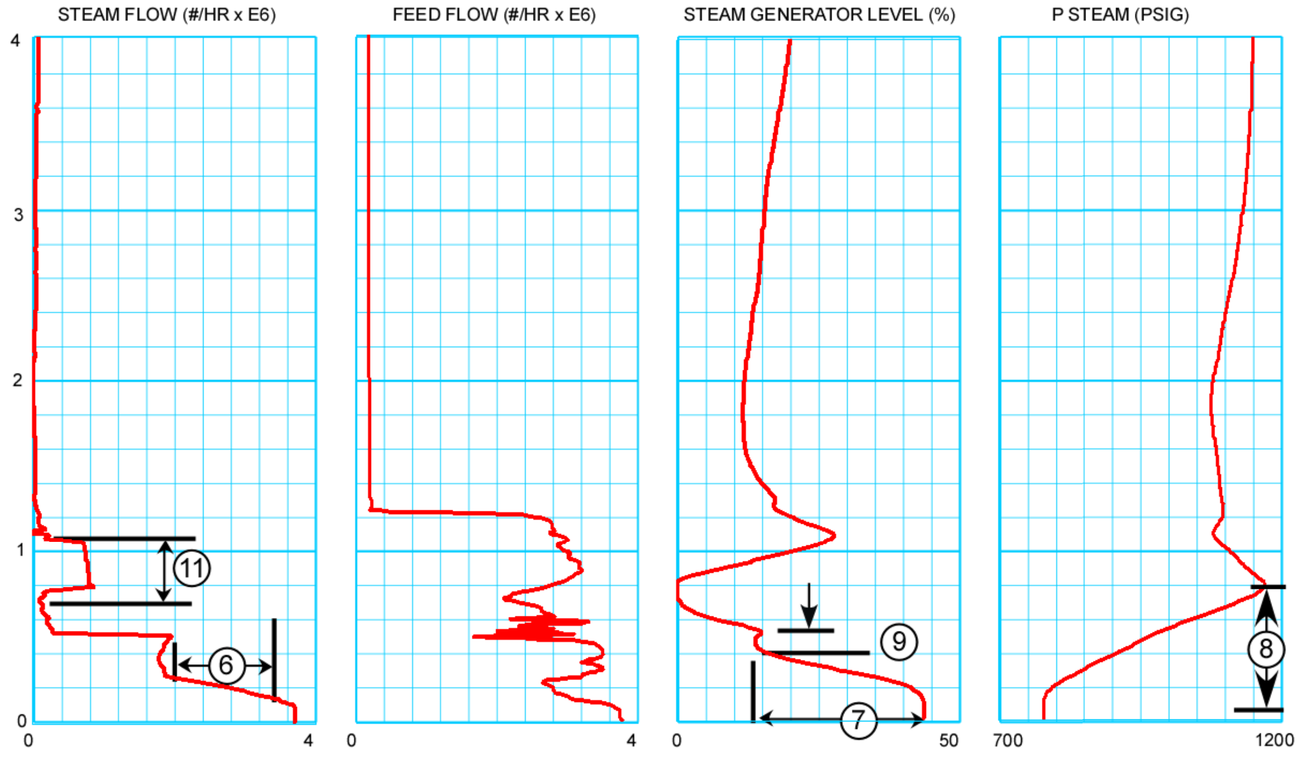
What this transient illustrates:

1. The only difference between this transient and transient 5.13 is that control rods are in manual. This makes almost no difference in the result. The trip occurs so quickly on high pressurizer pressure that no appreciable negative reactivity can be added.
2. It is the action of the steam dumps that make it possible to remain at power following a large load rejection.



Transient 5.14 Rapid Load Decrease, 100 - 50%, Steam Dumps Off, Rods In Manual

This page intentionally blank



TRANSIENT 5.14
RAPID LOAD DECREASE, 100-50%
STEAM DUMPS OFF, RODS IN MANUAL

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2231 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
Steam dumps turned off, rod bank selector switch placed in manual

Initiating Event:
Rapid closure of the turbine control valves with the valve position limiter

Transient 5.14 Rapid Load Decrease, 100 - 50%, Steam Dumps Off, Rods In Manual

This page intentionally blank

Transient 5.21 Dropped Rod (Shutdown Bank A Rod M-14)

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Shutdown bank A rod M-14 drops completely into the core

Point Explanation

1. **Nuclear power** drops in response to the negative reactivity insertion associated with the dropped rod. Of the two power range channels plotted, the power drop is greatest on channel NI-44 because it is closest to the dropped rod and measuring power in the most affected quadrant of the core. (Shutdown bank A rod M-14 is very close to the core "corner" adjacent to NI-44. See the attached core map.)

2. **Bank D rod position** increases with the power mismatch circuit of the rod control system calling for outward rod motion (nuclear power is decreasing relative to turbine load). The rod motion stops when bank D rods reach the automatic-rod-stop interlock setpoint of 223 steps.

3. **T_{avg} (all loops)** decreases with the power mismatch caused by the dropped rod, with nuclear power < turbine load. The decrease in T_{avg} continues until enough positive reactivity has been added from MTC to make nuclear power = to turbine load. (The drop in coolant temperature and the positive reactivity associated with it indicate that the positive reactivity associated with the rod withdrawal of point 2 above is not enough to counteract the negative reactivity of the dropped rod.) Note the spread in the loop temperatures: loop #3, closest to the dropped rod, is affected the most (its T_{avg} undergoes the greatest drop); loop #2 is next closest; loop #4 is third closest; and loop #1, which "flows" through the core quadrant diametrically opposite from the quadrant containing the dropped rod, is affected the least. (Don't be fooled by the relative T_{avg} values and the expanded scale: the loop #4 T_{avg} starts out and stays higher than the other three loop T_{avg}'s, but it decreases more than the loop #1 T_{avg}.) The temperature spread graphically illustrates that complete coolant mixing in the reactor and in the reactor vessel upper plenum does not occur.

4. The positive reactivity from the coolant temperature drop and (to a small extent) from the rod withdrawal increases **nuclear power**. The effect of MTC is most obvious in the quadrant measured by NI-43, which is diametrically opposite from the location of the dropped rod.

<u>**Point**</u> <u>**Explanation**</u>

5. **Steam pressure** drops as heat transfer conditions in the SGs change due to the reduced T_{avg}. The lower T_{avg} cannot continue to support steam pressure at its initial value. $\dot{Q} = UA(T_{avg} - T_{stm})$; $\dot{Q}$ is essentially constant with the turbine control valve position unchanging, so T_{stm} (and P_{stm}) is gradually decreasing to maintain the same ΔT across the SG tubes.

6. **Generator load** gradually falls off with the degraded steam pressure. Throttle pressure remains below the operating range of the throttle pressure compensator, so the control valves remain at their initial positions, and load becomes a function of steam pressure.

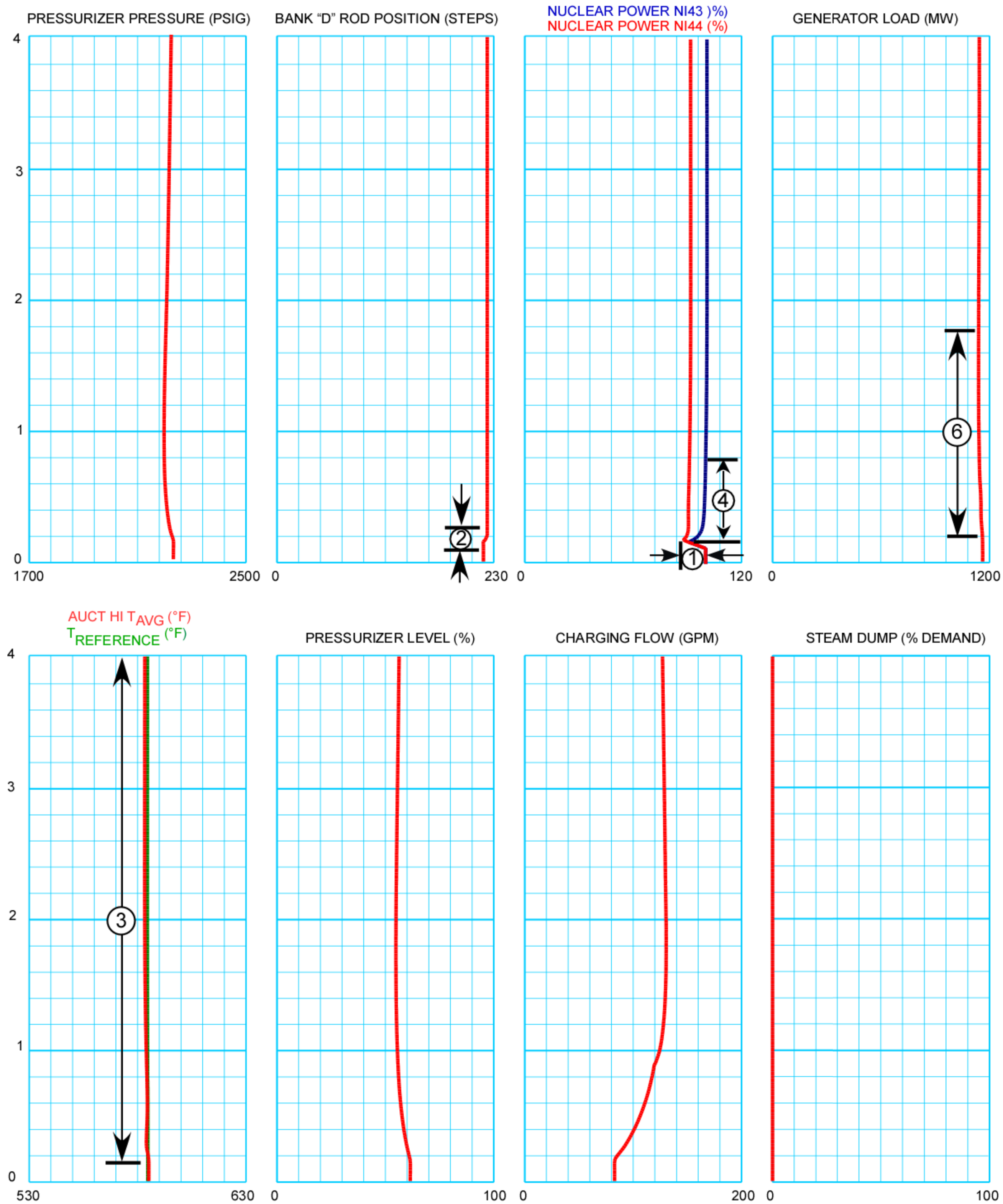
Note 1: The dropped rod is located closer to the edge of the core than any other rod, except for its symmetrical counterparts. At 100% power, any dropped rod closer to the center of the core causes a high negative flux rate trip. The core map on the MID panel in the simulator effectively illustrates the rod location.

Note 2: At steady state at the end of the transient, the negative reactivity from the dropped rod is balanced by the positive reactivity from the decrease in coolant temperature and from the slight decrease in fuel temperature associated with the slight decrease in power. The following summarizes the trends in many of the major plant parameters during this transient, steady state to steady state:

Auctioneered high T_{avg} drops from 585.1°F to 583.3°F.
Steam pressure drops from 792 psig to 770 psig.
Generator load drops from 1162 MW to 1143 MW (98.5%).
NI-44 power drops to 91.8%.
NI-43 power increases to 100.8%.

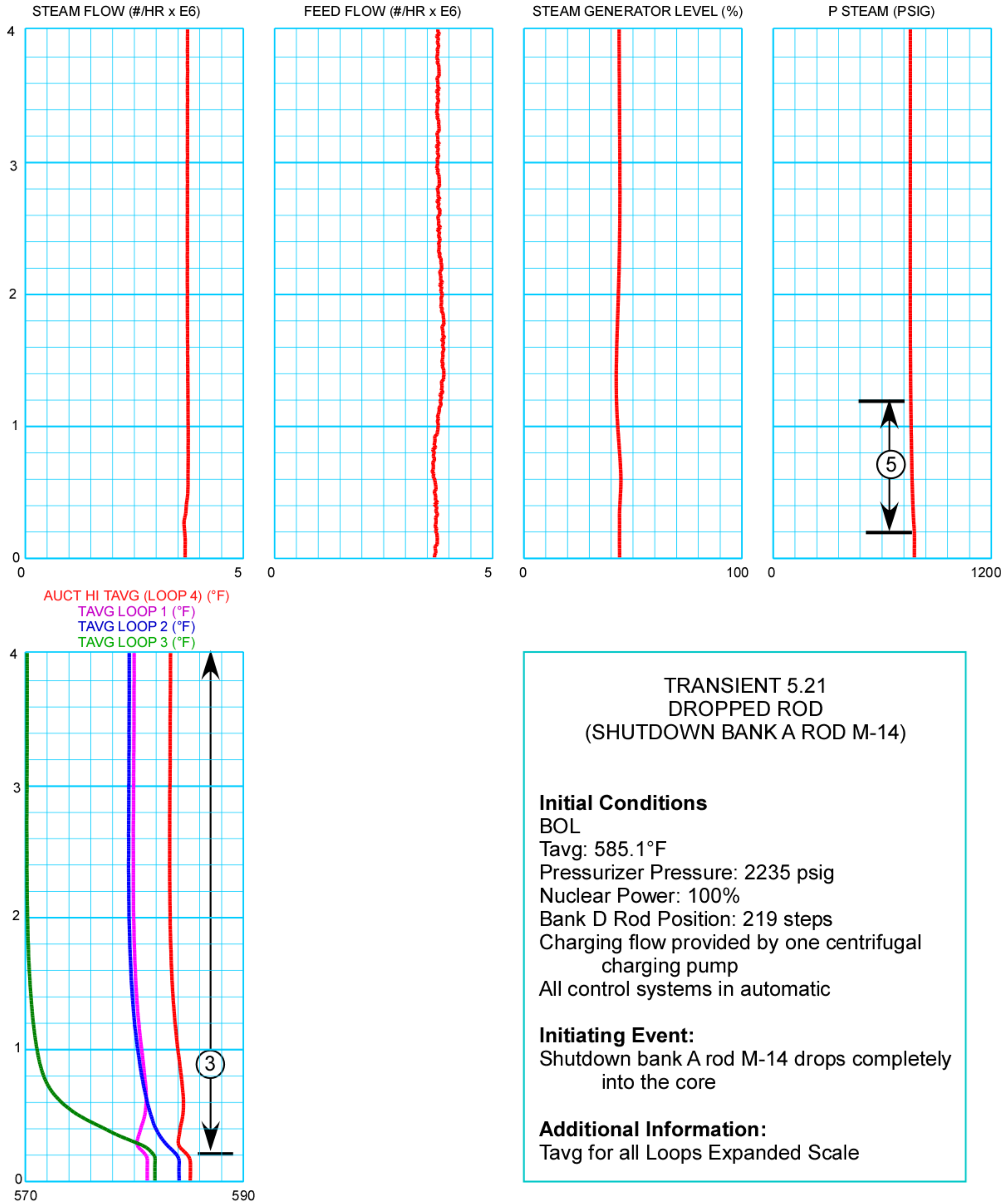
What this transient illustrates:

1. The negative reactivity associated with a dropped rod.
2. The spread in coolant temperatures between core quadrants, illustrating that the dropped rod affects the portions of the core closest to it the most.



Transient 5.21 Dropped Rod (Shutdown Bank A Rod M-14)

This page intentionally blank



TRANSIENT 5.21
DROPPED ROD
(SHUTDOWN BANK A ROD M-14)

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Shutdown bank A rod M-14 drops completely into the core

Additional Information:
Tavg for all Loops Expanded Scale

Transient 5.21 Dropped Rod (Shutdown Bank A Rod M-14)

This page intentionally blank

Transient 5.22 Fast Rod Withdrawal, 45% Load

Initial Conditions:

BOL	Bank D Rod Position: 103 steps
T_{avg}: 568.7°F	Nuclear Power: 45%
PZR Pressure: 2237 psig	All control systems in automatic

Initiating Event: Rod control system controller failure withdraws bank D rods at 72 steps/min

Point Explanation

1. **Bank D rod position** increases at 72 steps/min in response to the controller failure.

2. **Nuclear power** increases with the positive reactivity associated with the rod withdrawal. The increase in power is tempered somewhat by the negative reactivity associated with the increase in reactor coolant temperature.

3. $\mathbf{T_{avg}}$ increases with the power mismatch caused by the rod withdrawal, with nuclear power > turbine load. The increase in T_{avg} is also evident in the increase in pressurizer level.

4. **Steam pressure** increases as heat transfer conditions in the SGs change due to the increased T_{avg}. $\dot{Q} = UA(T_{avg} - T_{stm})$; $\dot{Q}$ is essentially constant with turbine load unchanging, so T_{stm} (and P_{stm}) is gradually increasing to maintain the same ΔT across the SG tubes.

5. **Steam dump demand** steadily increases to maximum with $T_{avg} >> T_{ref}$. Steam dump actuation does not occur (there is no change in steam flow at this point), as there is no loss-of-load arming signal.

6. **Pressurizer level** increases because of the decrease in reactor coolant density (T_{avg} is increasing).

7. Steam dump actuation brings $\mathbf{T_{avg}}$ toward no-load T_{avg}. The steam dumps are armed by the turbine trip, and the large initial demand (T_{avg} - no-load T_{avg}) causes all 12 steam dump valves to open. The valves modulate closed as the steam dump demand is decreased.

8. **Pressurizer pressure** rises as the steam bubble is squeezed by the thermal expansion of the reactor coolant. The pressure increase is initially turned by PORV actuation. The third rise in pressure is too severe to be mitigated by PORV operation, so the reactor trips on high pressurizer pressure (2385 psig). Pressure then drops rapidly with the expansion of the pressurizer steam bubble associated with the post-trip reactor coolant contraction. The decreased reactor coolant temperature caused by steam dump operation increases the coolant density.

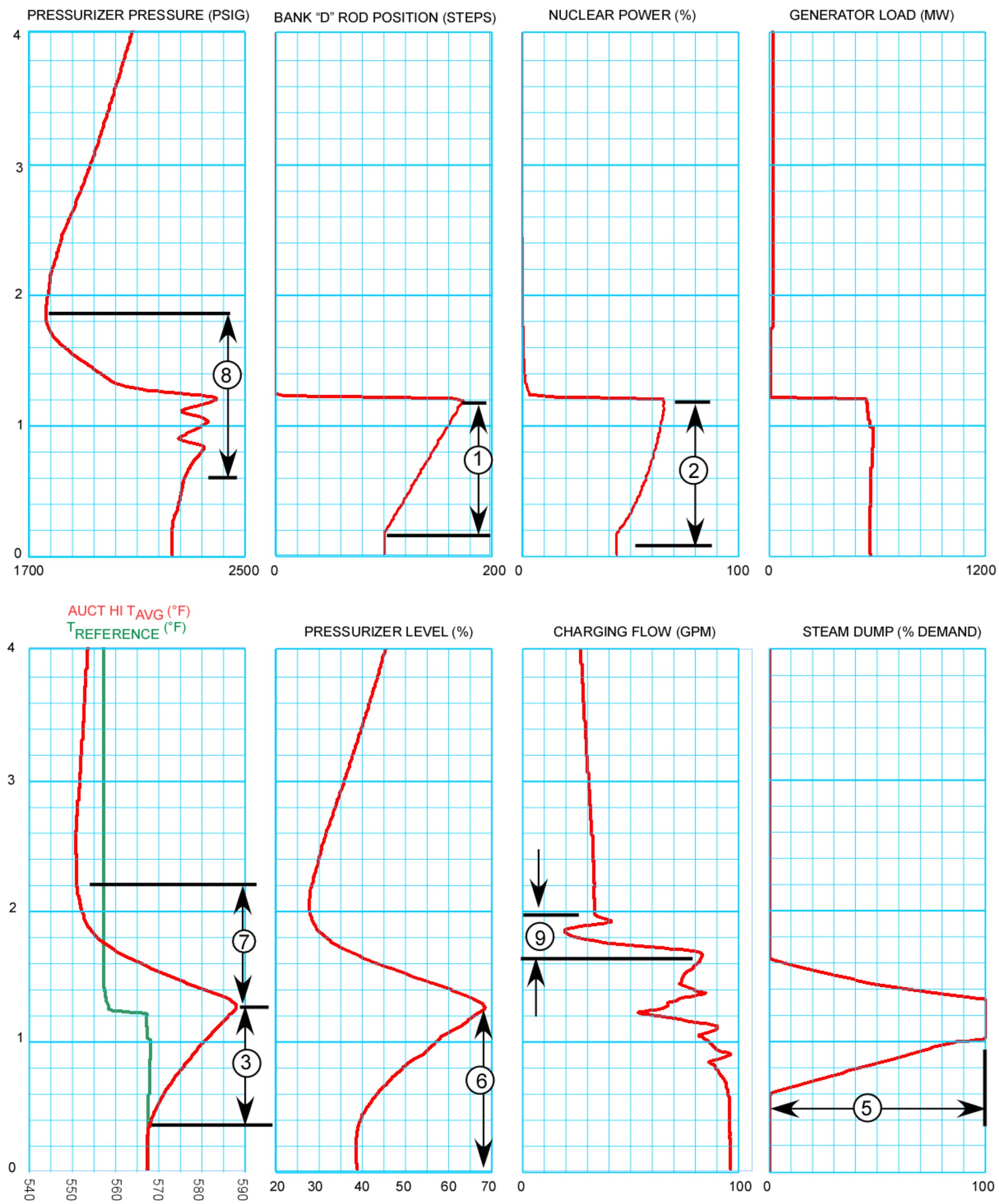
<u>**Point**</u> <u>**Explanation**</u>

9. **Charging flow** perturbation reflects an ESF actuation. The low pressurizer pressure ESF actuation is caused by the rapid expansion of the pressurizer steam bubble resulting from the rapid reactor coolant volume contraction following the trip.

Note 1: The "blip" in generator load at ~ 2 min reflects the voltage/current spike accompanying the opening of the output breakers.

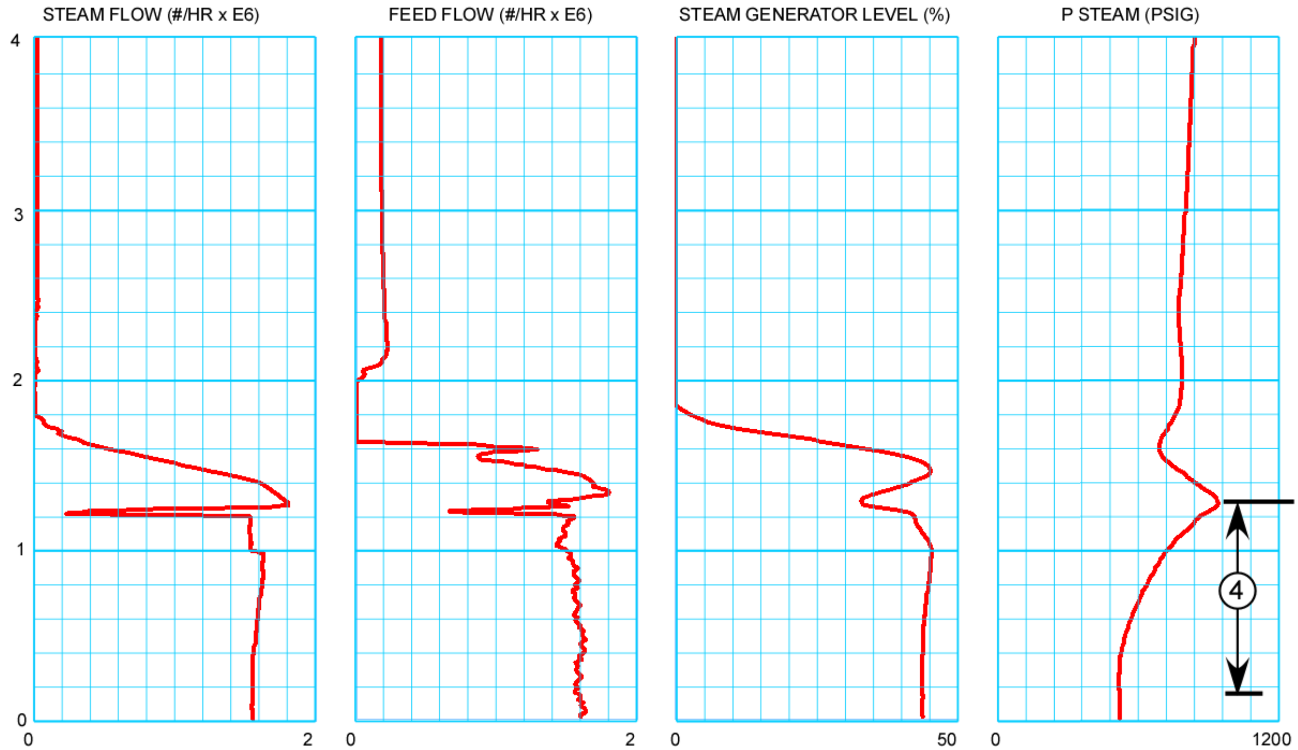
What this transient illustrates:

1. The response of the plant to a fast rod withdrawal at power, particularly the increases in reactor coolant temperature, pressurizer level, and steam pressure.
2. A high pressurizer pressure reactor trip.
3. A low pressurizer pressure ESF actuation.
4. The throttle pressure compensator keeps generator load fairly constant as steam pressure changes, since pressure remains in the operating band of the throttle pressure compensation circuit.



Transient 5.22 Fast Rod Withdrawal, 45% Load

This page intentionally blank



TRANSIENT 5.22
FAST ROD WITHDRAWAL, 45% LOAD

Initial Conditions
BOL
Tavg: 568.7°F
Pressurizer Pressure: 2252 psig
Nuclear Power: 45%
Bank D Rod Position: 103 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Rod control system controller failure withdraws bank D rods at 72 steps/min.

Transient 5.22 Fast Rod Withdrawal, 45% Load

This page intentionally blank

Transient 5.23 Fast Rod Withdrawal From Source Range

Initial Conditions:

BOL	Bank D Rod Position: 105 steps
T_{avg}: 558.1°F	Nuclear Power: 551 cps in S.R.
PZR Pressure: 2235 psig	Normal plant configuration for hot standby, except S.R. and I.R. trips are bypassed at NI cabinets

Initiating Event: Rod control system controller failure withdraws bank D rods at 72 steps/min

Point **Explanation**

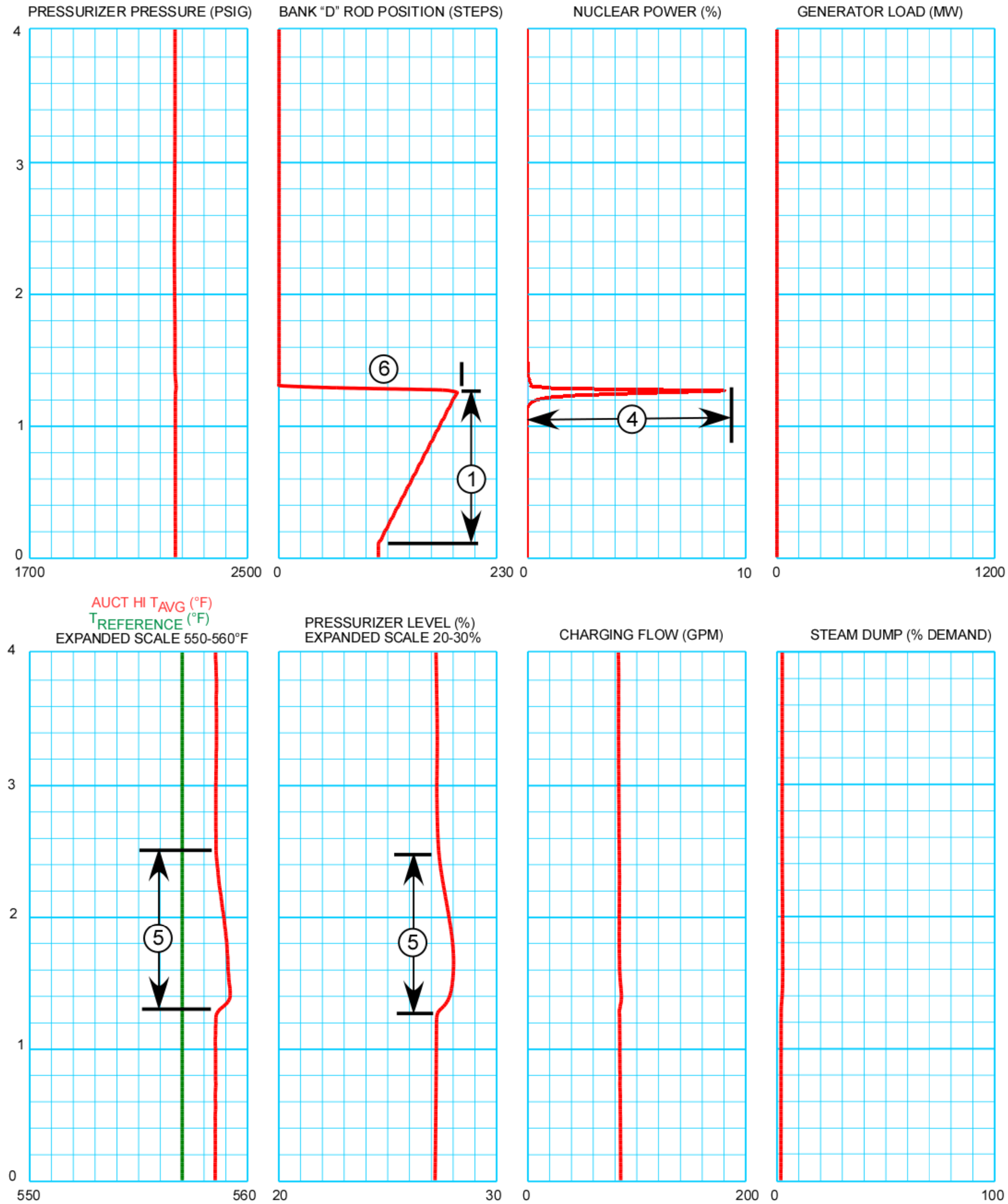
1. **Bank D rod position** increases at 72 steps/min in response to the controller failure.
2. **Source range power** increases with the positive reactivity associated with the rod withdrawal.
3. **Intermediate range power** also registers the increase in power caused by the rod withdrawal.
4. **Nuclear power (power range)** is on scale when rod withdrawal has increased neutron flux sufficiently. The point of adding heat has been reached.
5. **Pressurizer level** and **T_{avg}** increase slightly after the point of adding heat has been reached. Heat transfer from the fuel to the coolant is now great enough to raise coolant temperature.
6. The reactor trip (indicated by the step drop in **bank D rod position**) is caused by a high positive flux rate. The turbine trip and P-7 reactor trip is not a possible reactor trip, because the source range plot illustrates that the source range instruments have not been de-energized by P-10 at any time during the transient.

Note: The increase in reactor power caused by the rod withdrawal raises T_{avg} to a peak of 559.2°F. The increase in coolant temperature and corresponding increase in steam pressure do increase steam flow through the steam dump valves (the steam dumps are operating in the steam pressure mode of control), but the steam flow value is two orders of magnitude less than the range of indication and thus cannot be distinguished on the steam flow plot.

What this transient illustrates:

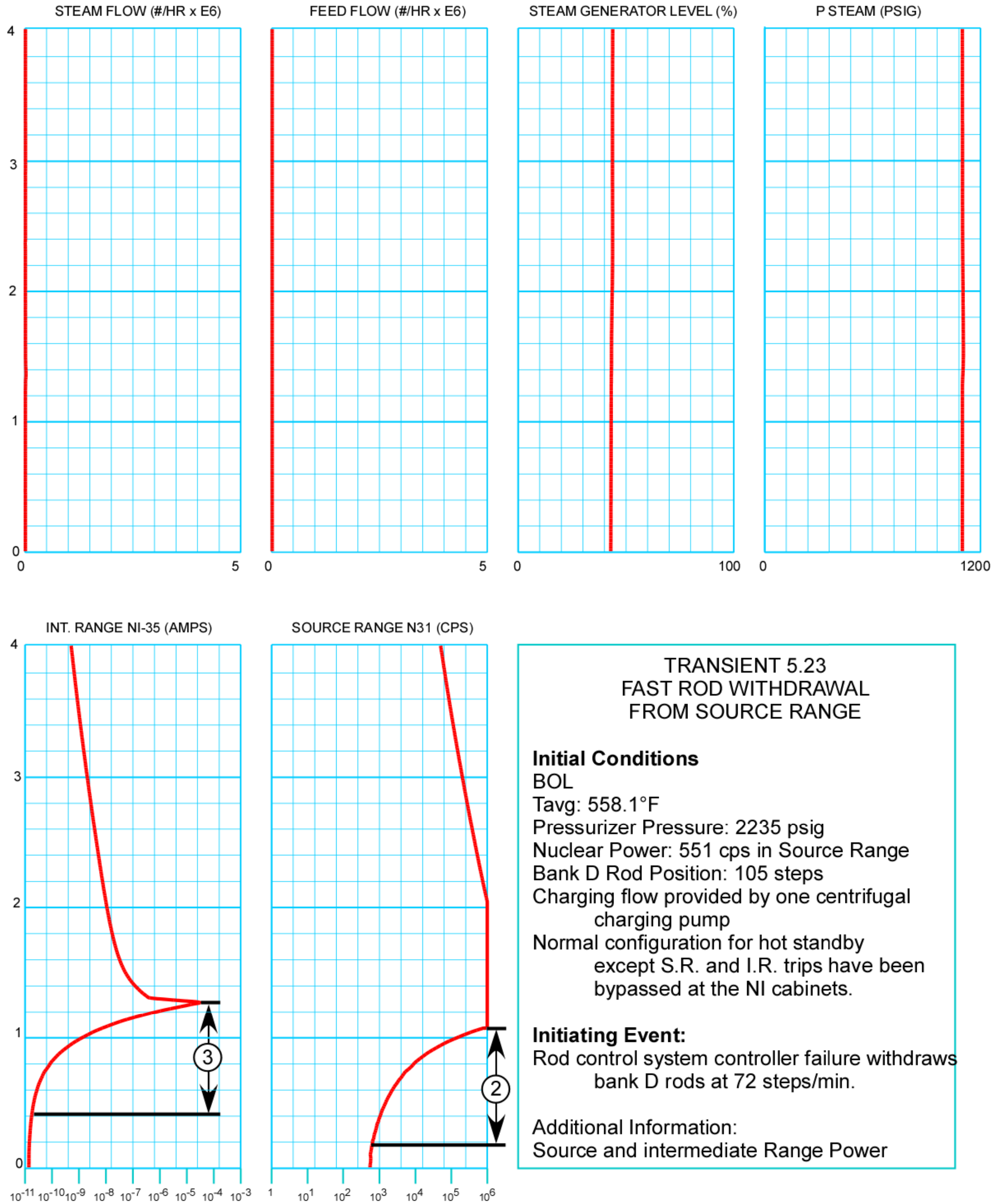
1. The response of the plant to a fast rod withdrawal starting in the source range, particularly the increase in power as measured by all three ranges of excore instruments.
2. A high positive flux rate reactor trip.

This page intentionally blank



Transient 5.23 Fast Rod Withdrawal From Source Range

This page intentionally blank



Transient 5.23 Fast Rod Withdrawal From Source Range

This page intentionally blank

Transient 5.31 Loop #1 Cold-Leg RTD Fails High

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.2°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	All control systems in automatic

Initiating Event: Loop #1 cold-leg RTD fails high

Point **Explanation**

1. **Auctioneered high T_{avg}** undergoes a rapid increase to > 620°F. The loop #1 T_{avg} becomes auctioneered high T_{avg}; that loop has a T_c of 630°F (the upper limit of the instrument range) because of the failed-high cold-leg RTD and a T_H of > 610°F (normal).

2. **Bank D rod position** decreases at the maximum rate (72 steps/min) because of the large temperature mismatch input to the rod control system calling for fast rod insertion. The large temperature mismatch results from the failed-high auctioneered high T_{avg}.

3. **Nuclear power** decreases due to the negative reactivity associated with rod insertion. The positive reactivity insertion resulting from the decrease in reactor coolant temperature, as discussed in point 4 below, is not enough to counteract it.

4. **Actual T_{avg}** decreases due to the large imbalance between nuclear power and secondary load, with nuclear power decreasing rapidly. To maintain secondary load, energy must be taken from the reactor coolant, reducing its temperature.

5. The **pressurizer level** decrease reflects the reduction in reactor coolant volume caused by the decrease in coolant temperature.

6. **Charging flow** increases greatly with pressurizer level low relative to the level setpoint, which remains at the maximum value (61.5%) with auctioneered high T_{avg} > 584.7°F.

7. **Steam dump** demand increases to 100% with the RTD failure. The $T_{avg} - T_{ref}$ input to the loss-of-load controller is maximized with $T_{avg} >> T_{ref}$. Note that at this time the dumps have not actuated, as there is no loss-of-load arming signal (no drop in turbine load yet).

8. **Pressurizer pressure** drifts lower as the pressurizer steam bubble expands with reactor coolant contraction. The pressurizer heaters cannot maintain normal operating pressure.

9. **Steam pressure** drops as heat transfer conditions in the SGs change due to the reduced T_{avg}. The lower T_{avg} cannot continue to support steam pressure at its initial value. $\dot{Q} = UA(T_{avg} - T_{stm})$; $\dot{Q}$ is essentially constant with the turbine control valve position unchanging, so T_{stm} (and P_{stm}) is gradually decreasing to maintain the same ΔT across the SG tubes.

<u>Point</u> **<u>Explanation</u>**

10. **Generator load** gradually falls off with the degraded steam pressure because throttle pressure remains below the range of the throttle pressure compensation circuit. Note: the faster dropoff in load just prior to the turbine trip reflects the response of the initial pressure limiter, a GE turbine EHC system feature which closes the control valves in response to a throttle pressure that decreases to less than 90% of a manually adjustable setpoint.

11. **Steam flow** increases sharply at approximately 1 min, 30 sec. The reduction in turbine load discussed in point 10 arms the dumps on the loss-of-load controller, and the existing 100% demand causes all 12 dump valves to blast open. The steam flow increase results.

12. The reactor trip (indicated by the step drop in **bank D rod position**) is caused by an ESF actuation on high steam flow plus low steam pressure. The high steam flow setpoint is reached with the steam dump valves blasting open, and the low steam pressure setpoint is reached as the steam pressure constantly decreases. Note that the ESF actuation takes place when the steam pressure is ~ 685 psig; the input to the low steam pressure bistable reaches the low steam pressure setpoint before actual pressure does because of the lead-lag circuit through which the steam pressure signal is processed. Note also that the other ESF actuation signals are not possible: pressurizer pressure and T_{avg} are not low enough yet, there is no steam line ΔP, and there is nothing causing containment pressure to increase.

13. The **charging flow** perturbation reflects the ESF actuation. The initial drop in flow reflects charging line isolation; the flow settles at the relatively high value governed by the maximum opening of charging flow control valve FCV-121 (the pressurizer level is low throughout most of the transient) and the flow restriction of the RCP seals (the only path left for injection from the CVCS).

14. **Pressurizer level** recovers with ECCS flow and letdown isolation following the ESF actuation.

15. **Feed flow** after the trip reflects AFW system operation (main feedwater is now isolated).

16. **Steam flow** decreases to 0, reflecting the MSIV closure associated with the high steam flow ESF actuation.

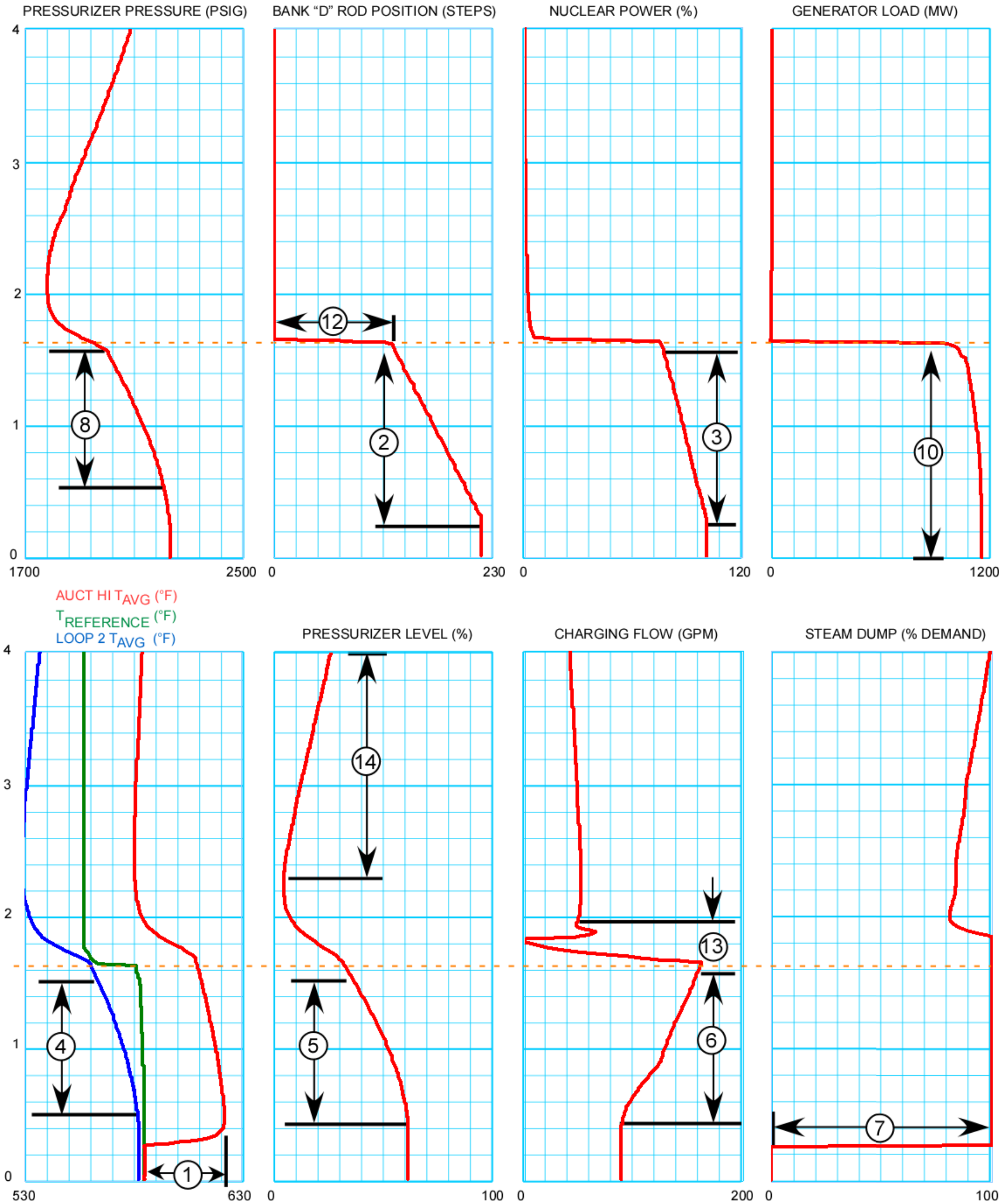
Note 1: The "blip" in generator load shortly after 2 min reflects the voltage/current spike accompanying the opening of the output breakers.

Note 2: For pt. 12, the lead-lag unit changes a ramp of f(t) = kt into $k[t + (\tau_1 - \tau_2)(1 - e^{-t/\tau_2})]$ for a lead constant of τ_1 and a lag constant of τ_2. For the low steam pressure portion of the high steam flow ESF actuation, τ_1 = 50 sec, τ_2 = 5 sec. So, after the exponential term dies out, ramp kt becomes k(t + 45 sec).

What this transient illustrates:

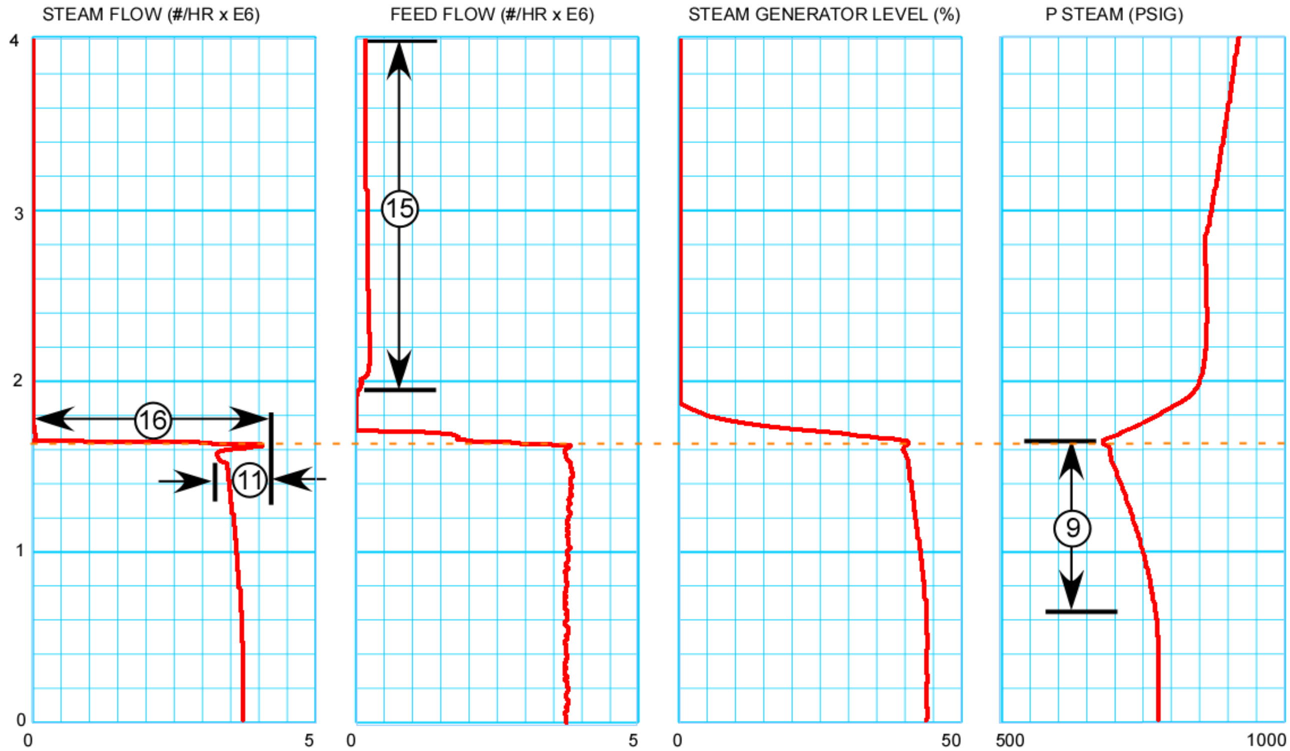
1. The responses of the rod control, pressurizer level control, and steam dump control systems to a failed-high cold-leg RTD.
2. The decrease in T_{avg} when nuclear power << secondary load.
3. A reactor trip and ESF actuation on high steam flow + low steam pressure.

This page intentionally blank



Transient 5.31 Loop #1 Cold-Leg RTD Fails High

This page intentionally blank



TRANSIENT 5.31
LOOP #1 COLD-LEG RTD FAILS HIGH

Initial Conditions
BOL
Tavg: 585.2°F
Pressurizer Pressure: 2245 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Loop #1 cold-leg RTD fails high

Transient 5.31 Loop #1 Cold-Leg RTD Fails High

This page intentionally blank

Transient 5.32 Loop #1 Hot-Leg RTD Fails High, ~ 25% Load

Initial Conditions:

BOL	Bank D Rod Position: 187 steps
T_{avg}: 562.7°F	Nuclear Power: 32%
PZR Pressure: 2235 psig	Steam dumps armed due to previous load rejection

Initiating Event: Loop #1 hot-leg RTD fails high

Point **Explanation**

1. **Auctioneered high T_{avg}** increases rapidly to ~ 600°F. The loop #1 T_{avg} becomes auctioneered high T_{avg}; that loop has a T_H of 650°F (the upper limit of the instrument range) because of the failed hot-leg RTD, an increase of some 80°F above its initial value. This increase in T_H increases T_{avg} by ~ 40°F.
2. **Charging flow** increases greatly with pressurizer level low relative to the level setpoint, which has increased to the maximum value (61.5%) with auctioneered high T_{avg} > 584.7°F.
3. **Steam dump demand** increases to 100% with the RTD failure. The T_{avg} - T_{ref} input to the loss-of-load controller is maximized with $T_{avg} >> T_{ref}$. As discussed in point 4 below, the dump valves open immediately, as the steam dump control system is already armed in the T_{avg} mode.
4. **Steam flow** increases sharply after the first several sec. With the dumps already armed and a 100% demand, all 12 dump valves blast open.
5. **Steam pressure** decreases sharply with all 12 dump valves opening and relieving steam to the condenser.
6. The reactor trip (indicated by the step drop in **bank D rod position**) is caused by an ESF actuation on high steam flow plus low steam pressure. The high steam flow setpoint is reached with the steam dump valves blasting open, and the low steam pressure setpoint is reached as the steam pressure constantly decreases. Note that the ESF actuation takes place when the steam pressure is ~ 887 psig. The low steam pressure bistable reaches the low steam pressure setpoint before actual pressure does because of the lead-lag circuit through which the steam pressure signal is processed. Figure 5.32-1 illustrates the effect of the Lead/Lag compensation. Note also that the other ESF actuation signals are not possible: pressurizer pressure and T_{avg} are not low enough yet, there is no steam line ΔP, and there is nothing causing containment pressure to increase.

Transient 5.32 Loop #1 Hot-Leg RTD Fails High, ~ 25% Load (cont'd)

Point Explanation

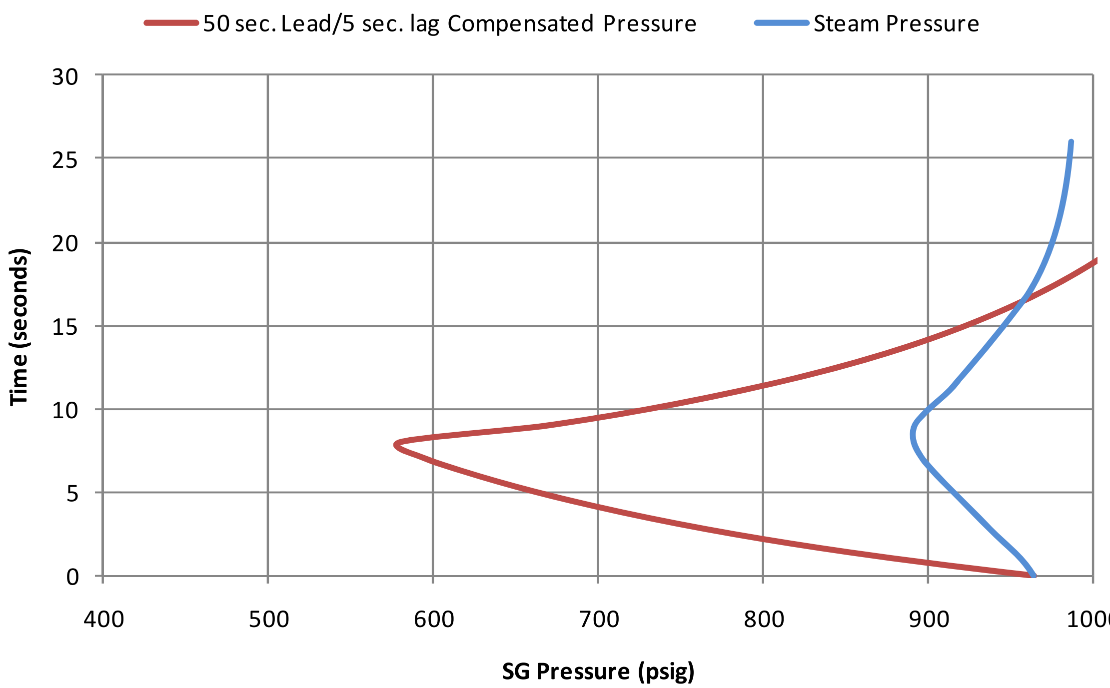


Figure 5.32-1, Effect of Lead/Lag Compensation

7. The **charging flow** perturbation reflects the ESF actuation. The initial drop in flow reflects charging line isolation; the flow settles at the relatively high value governed by the maximum opening of charging flow control valve FCV-121 (the pressurizer level is below the setpoint level [61.5%] throughout the transient) and the flow restriction of the RCP seals (the only path left for injection from the CVCS).

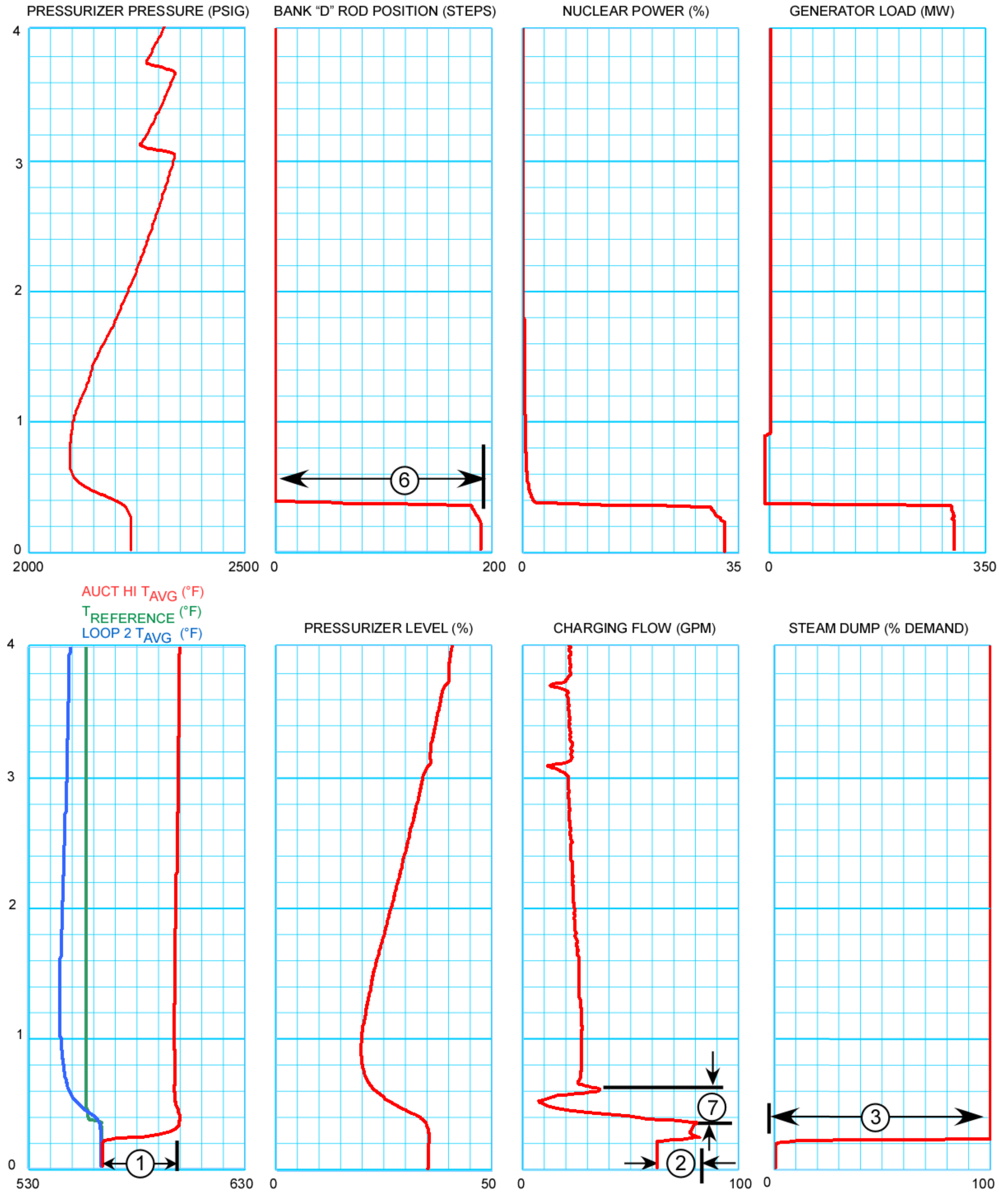
8. **Steam flow** decreases to 0, reflecting the MSIV closure associated with the high steam flow ESF actuation.

Note 1: The "blip" in generator load at ~ 48 sec reflects the voltage/current spike accompanying the opening of the output breakers.

What this transient illustrates:

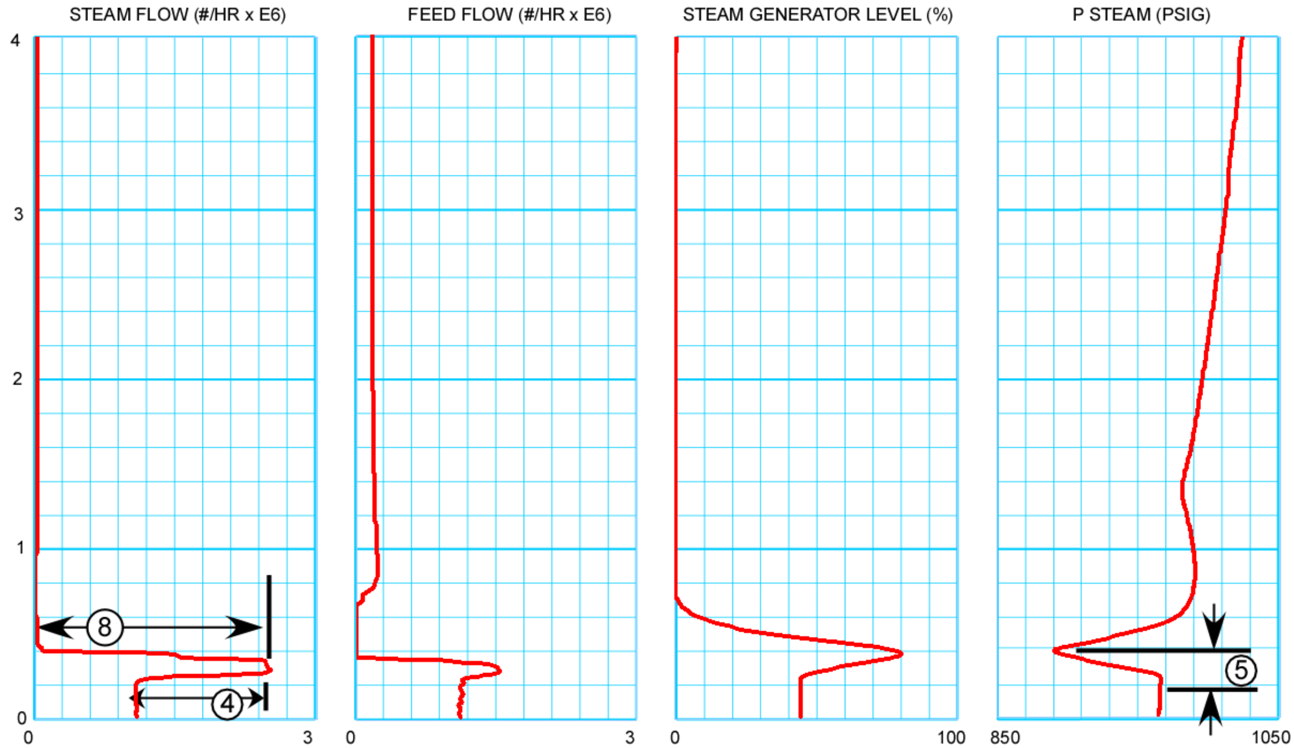
1. The responses of the pressurizer level control and (already armed) steam dump control systems to a failed-high hot-leg RTD.

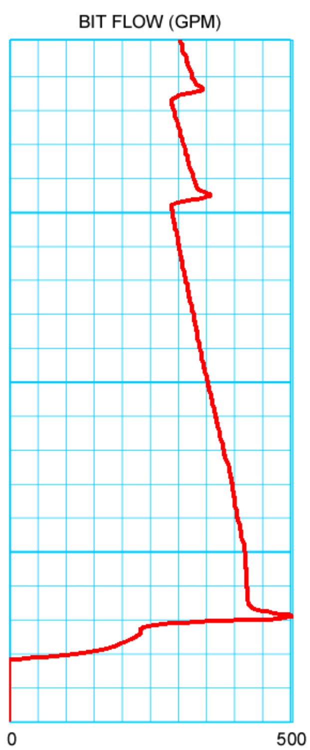
2. A reactor trip and ESF actuation on high steam flow + low steam pressure.



Transient 5.32 Loop #1 Hot-Leg RTD Fails High, ~ 25% Load

This page intentionally blank





TRANSIENT 5.32
LOOP #1 HOT-LEG RTD FAILS HIGH,
25% LOAD

Initial Conditions
BOL
Tavg: 562.7°F
Pressurizer Pressure: 2250 psig
Nuclear Power: 32%
Bank D Rod Position: 187 steps
Charging flow provided by one centrifugal charging pump
Steam dumps armed due to previous load rejection

Initiating Event:
Loop #1 hot-leg RTD fails high

Transient 5.32 Loop #1 Hot-Leg RTD Fails High, ~ 25% Load

This page intentionally blank

Transient 5.33 Power Range Channel NI-41 Fails High

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Power range channel NI-41 fails high

<u>Point</u> **<u>Explanation</u>**

1. **Nuclear power (from NI-41)** increases to maximum (initiating event).

2. **Bank D rod position** decreases at the maximum rate (72 steps/min). The failed-high power range channel becomes auctioneered high nuclear power, and its step increase relative to a constant turbine load develops a large rod insertion signal in the power mismatch circuit of the rod control system.

3. Actual **nuclear power** decreases due to the negative reactivity associated with rod insertion. The positive reactivity insertion resulting from the decrease in reactor coolant temperature, as discussed in point 4 below, is not enough to counteract it during this phase of the transient.

4. $\mathbf{T_{avg}}$ decreases due to the large imbalance between nuclear power and turbine load, with nuclear power decreasing rapidly. To maintain turbine load, energy must be taken from the reactor coolant, reducing its temperature. The reduction in T_{avg} is also evident in the decreasing pressurizer level and pressurizer pressure.

5. **Steam pressure** drops as heat transfer conditions in the SGs change due to the reduced T_{avg}. The lower T_{avg} cannot continue to support steam pressure at its initial value. $\dot{Q} = UA(T_{avg} - T_{stm})$; $\dot{Q}$ is essentially constant with the turbine control valve position unchanging, so T_{stm} (and P_{stm}) is gradually decreasing to maintain the same ΔT across the SG tubes.

6. **Generator load** gradually falls off with the degraded steam pressure. Throttle pressure remains below the range of the throttle pressure compensation circuit, so the control valves remain at their initial positions, and load becomes a function of steam pressure.

7. **Bank D rod position** stops decreasing. Over the first minute or so of the transient, the large negative input to the rod control system from the initial "spike" in auctioneered high nuclear power has been decaying off, while a large positive input due to the decrease in auctioneered high T_{avg} relative to T_{ref} has been building in. These inputs essentially cancel at ~ 1 min, and rod motion stops. For several seconds prior to this point, the rod insertion speed is slowing, showing that the total error calling for rod insertion has decreased greatly from the initial error associated with the NI-41 failure.

Point Explanation

8. **Bank D rod position** remains constant over the last three min of the transient. Ordinarily, outward rod motion would result from the continued decrease in T_{avg} and the continued decay of the nuclear power spike, but the failed channel exceeds the overpower rod stop setpoint (103%), and this rod stop requires only a one-out-of-four coincidence.

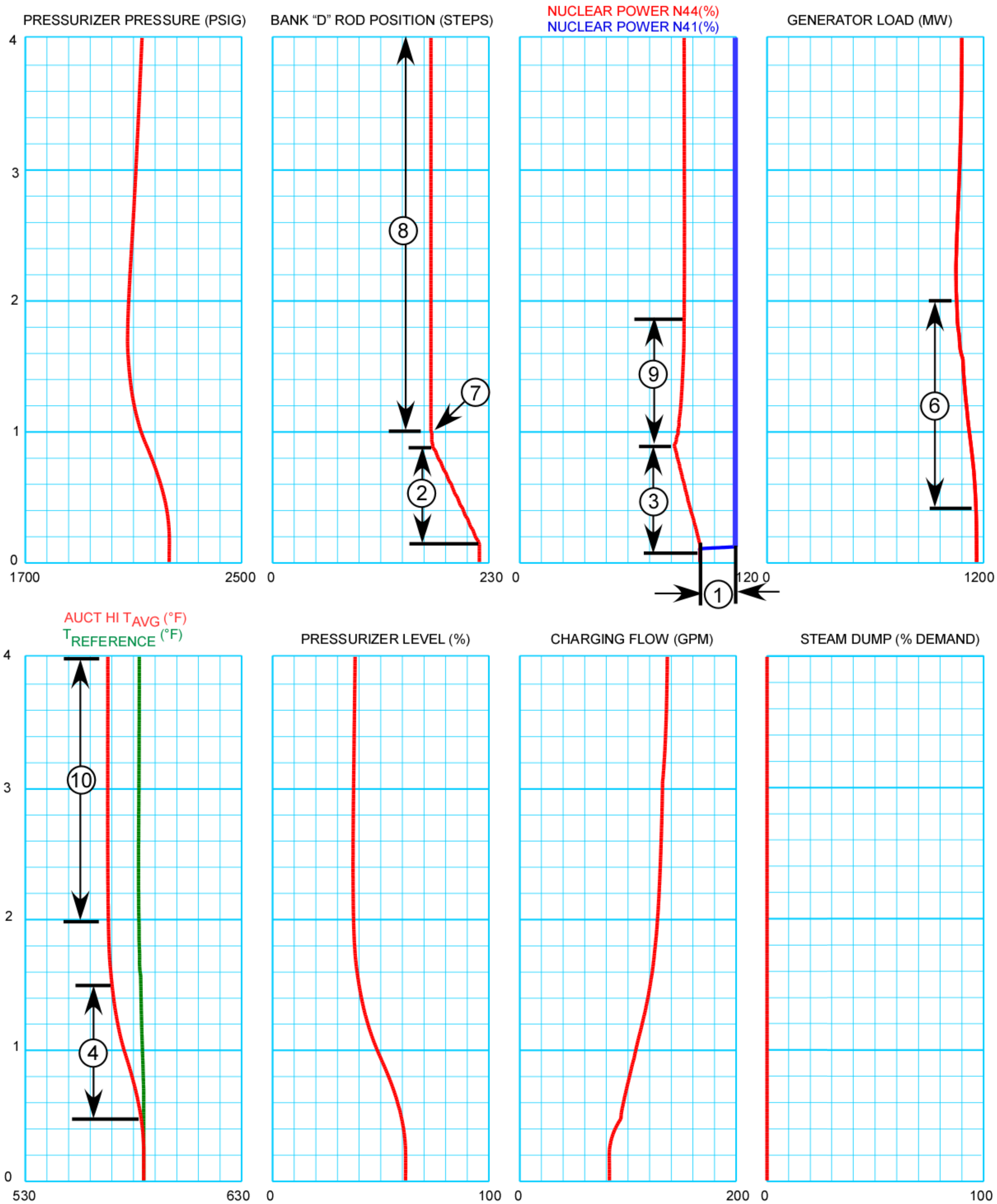
9. **Nuclear power** increases with the positive reactivity insertion associated with the reduction in coolant temperature after rod insertion stops.

10. $\mathbf{T_{avg}}$ remains constant over the final two min of the transient. At this point, nuclear power and turbine load are essentially equal, meaning that a power mismatch that would change T_{avg} no longer exists.

Note: By the end of the 4 plotted min, new steady-state conditions have essentially been reached: the negative reactivity from the rod insertion is balanced by the positive reactivity from the reduction in fuel temperature associated with the power change and from the reduction in coolant temperature, and T_{avg} is constant with nuclear power equal to the turbine load, which has decreased from its initial 100% value because of the degraded steam pressure. Final steady-state values: nuclear power = 91.2%, generator load = 1079 MW, T_{avg} = 568.2°F.

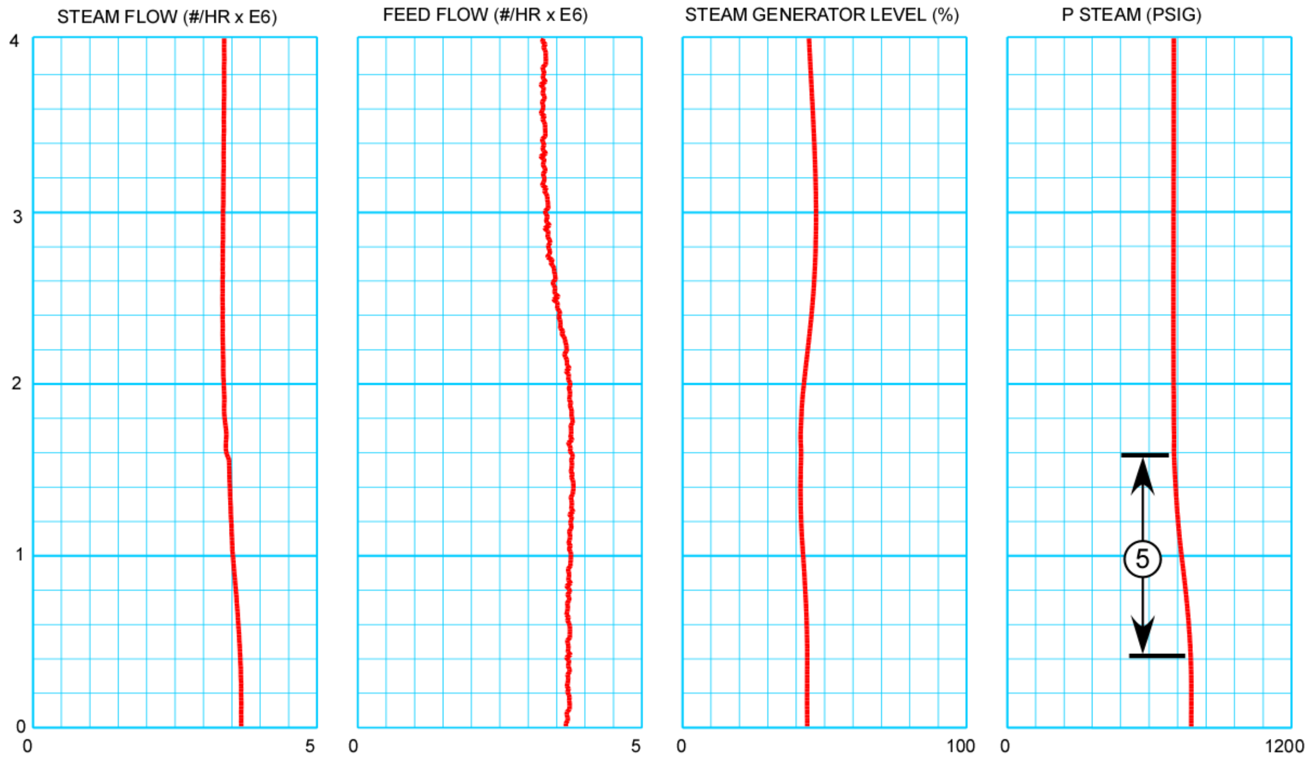
What this transient illustrates:

1. The response of the rod control system to a failed-high power-range NI.
2. The decrease in T_{avg} when nuclear power < secondary load.
3. An overpower rod stop.
4. Steady-state to steady-state reactivity balances with changing parameters.



Transient 5.33 Power Range Channel NI-41 Fails High

This page intentionally blank



TRANSIENT 5.33
POWER RANGE CHANNEL NI-41
FAILS HIGH

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Power range channel NI-41 fails high

Transient 5.33 Power Range Channel NI-41 Fails High

This page intentionally blank

Transient 5.34 Steam Dump Loss-Of-Load Controller Fails To Maximum Demand

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2233 psig	Steam dumps armed due to previous load rejection

Initiating Event: Steam dump loss-of-load controller fails to maximum demand

Point **Explanation**

1. The initiating event increases **steam dump demand** to 100%.

2. **Steam flow** increases sharply as the steam dump valves open. The dumps are already armed, and the 100% demand resulting from the controller failure causes all 12 dump valves to blast open.

3. **Steam generator level** swells as the steam dump valves open.

4. **Steam pressure** drops with the additional steam release associated with steam dump operation.

5. **Generator load** gradually falls off with the degraded steam pressure. Throttle pressure remains below the range of the throttle pressure compensation circuit, so the control valves remain at their initial positions, and load becomes a function of steam pressure.

6. The reactor trip (indicated by the step drop in **bank D rod position**) is caused by an ESF actuation on high steam flow plus low steam pressure. The high steam flow setpoint is reached with the steam dump valves blasting open, and the low steam pressure setpoint is reached as the steam pressure constantly decreases. Note that the ESF actuation takes place when the steam pressure is ~ 750 psig; the input to the low steam pressure bistable reaches the low steam pressure setpoint before actual pressure does because of the lead-lag circuit through which the steam pressure signal is processed. Note also that the other ESF actuation signals are not possible: pressurizer pressure and T_{avg} are not low enough yet, there is no steam line ΔP, and there is nothing causing containment pressure to increase.

7. The **charging flow** perturbation reflects the ESF actuation. The initial drop in flow reflects charging line isolation; the flow then returns to a relatively high value governed by the maximum opening of charging flow control valve FCV-121 (the pressurizer level is low immediately after the ESF actuation) and the flow restriction of the RCP seals (the only path left for injection from the CVCS).

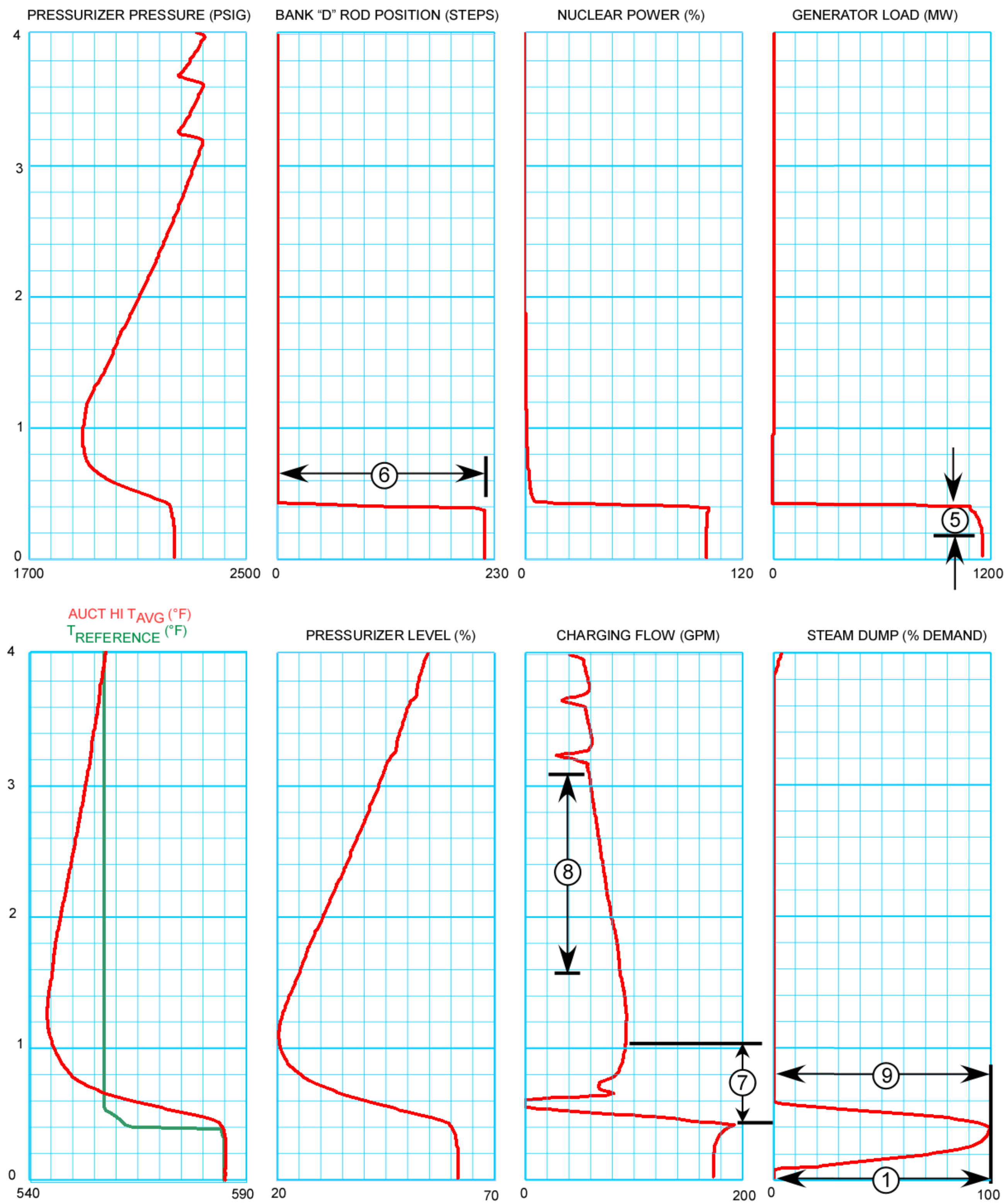
Point Explanation

8. **Charging flow** gradually decreases as pressurizer pressure rises due to ECCS injection compressing the steam bubble.

9. **Steam dump demand** returns to 0 after the turbine trip. The turbine trip which accompanies the reactor trip puts the turbine trip controller in play, and the steam dump demand falls as T_{avg} decreases to < the no-load value. **Note:** This transient is very similar to transient 5.32 (Loop #1 Hot-Leg RTD Fails High, 25% Load), transient 5.3A (Loop #1 Hot-Leg RTD Fails High, 100%), and transient 5.3B (Loop #1 Cold-Leg RTD Fails High, 100%). In those transients, the already armed steam dump control system opens all steam dump valves when auctioneered high T_{avg} fails high, which results in a large steam dump demand.

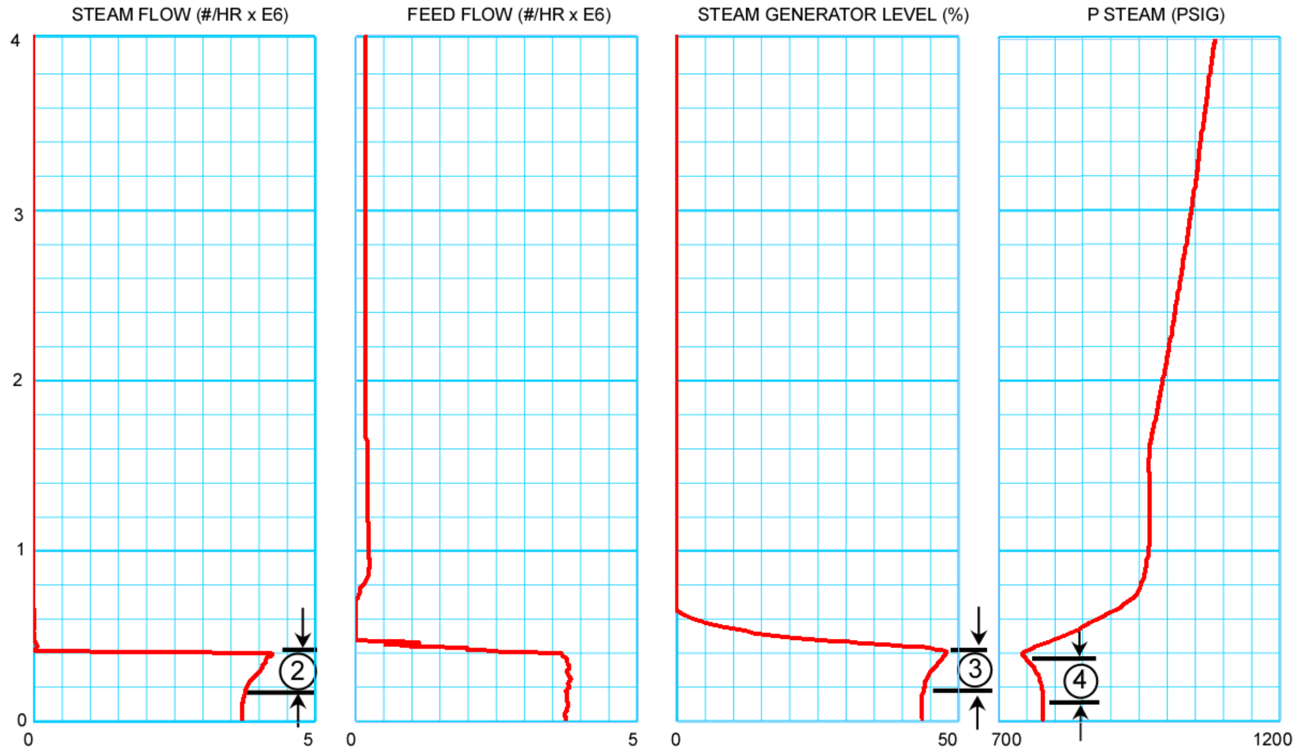
What this transient illustrates:

1. The response of the (already armed) steam dump control system to a loss-of-load controller demand failure.
2. A reactor trip and ESF actuation on high steam flow + low steam pressure.



Transient 5.34 Steam Dump Loss-Of-Load Controller Fails To Maximum Demand

This page intentionally blank



TRANSIENT 5.34
STEAM DUMP LOSS-OF-LOAD CONTROLLER
FAILS TO MAXIMUM DEMAND

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Steam dumps armed from previous load rejection
All control systems in automatic

Initiating Event:
Steam dump loss-of-load controller fails to maximum demand

Transient 5.34 Steam Dump Loss-Of-Load Controller Fails To Maximum Demand

This page intentionally blank

Transient 5.35 Impulse Pressure Channel Pt-505 Fails Low

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.2°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	All control systems in automatic

Initiating Event: Impulse pressure channel PT-505 fails low

Note: This impulse pressure channel feeds the rod control system (T_{ref} for the temperature mismatch circuit and P_{imp} for the power mismatch circuit), the steam dump control system (T_{ref} for the loss-of-load controller), and the steam generator water level control system (the level setpoint varies linearly with P_{imp} from 0 - 20% load); provides one input to the P-13 permissive (turbine at-power permissive); establishes the high steam flow setpoint for the high steam flow + low-low T_{avg} or low steam pressure ESF actuation in protection channel I; and supplies the C-5 interlock (automatic rod withdrawal block < 15% load).

Point **Explanation**

1. **T_{ref}** undergoes a rapid decrease to its minimum value of 557°F (programmed T_{avg} for no load).
2. **Bank D rod position** decreases at the maximum rate (72 steps/min) because of the large temperature mismatch (T_{ref} << T_{avg}) and large power mismatch (turbine load decreasing rapidly relative to nuclear power) inputs to the rod control system calling for fast rod insertion. The large mismatches result from the failed-low impulse pressure channel.
3. **Nuclear power** decreases due to the negative reactivity associated with the rod insertion. The positive reactivity insertion resulting from the decrease in reactor coolant temperature, as discussed in point 4 below, is not enough to counteract it.
4. **T_{avg}** decreases due to the large imbalance between nuclear power and turbine load, with nuclear power decreasing rapidly. To maintain turbine load, energy must be taken from the reactor coolant, reducing its temperature.
5. The **pressurizer level** decrease reflects the reduction in reactor coolant volume caused by the decrease in coolant temperature.
6. **Charging flow** increases greatly with pressurizer level low relative to the level setpoint. Both the pressurizer level and the level setpoint (a function of auctioneered high T_{avg}) are decreasing, but the pressurizer level remains below the level setpoint.
7. **Steam dump demand** increases to 100% with the impulse pressure failure. The T_{avg} - T_{ref} input to the loss-of-load controller is maximized with T_{avg} >> T_{ref}. Note that at this time the dumps have not actuated, as there is no loss-of-load arming signal (no drop in turbine load yet).

<u>Point</u> <u>Explanation</u>

8. **Pressurizer pressure** drifts lower as the pressurizer steam bubble expands with reactor coolant contraction. The pressurizer heaters cannot maintain normal operating pressure.

9. **Steam pressure** drops as heat transfer conditions in the SGs change due to the reduced T_{avg}. The lower T_{avg} cannot continue to support steam pressure at its initial value. $\dot{Q} = UA(T_{avg} - T_{stm})$; $\dot{Q}$ is essentially constant with the turbine control valve position unchanging, so T_{stm} (and P_{stm}) is gradually decreasing to maintain the same ΔT across the SG tubes.

10. **Generator load** gradually falls off with the degraded steam pressure, and, after ~ 1 min 24 sec, with throttling of the turbine control valves in response to the initial pressure limiter. Throttle pressure remains below the range of the throttle pressure compensation circuit, so at first the control valves remain at their initial positions, and load becomes a function of steam pressure. Later, the initial pressure limiter closes the control valves in response to a throttle pressure that has decreased to less than 90% of setpoint.

11. **Steam flow** decreases first with the reduction in steam pressure and later with closure of the turbine control valves by the initial pressure limiter.

12. **Steam generator level** decreases with the reduction in feed flow (the failed-low impulse pressure channel has driven the steam generator water level setpoint to its minimum programmed value of 33%, so feed flow is decreased to bring level down to the new setpoint).

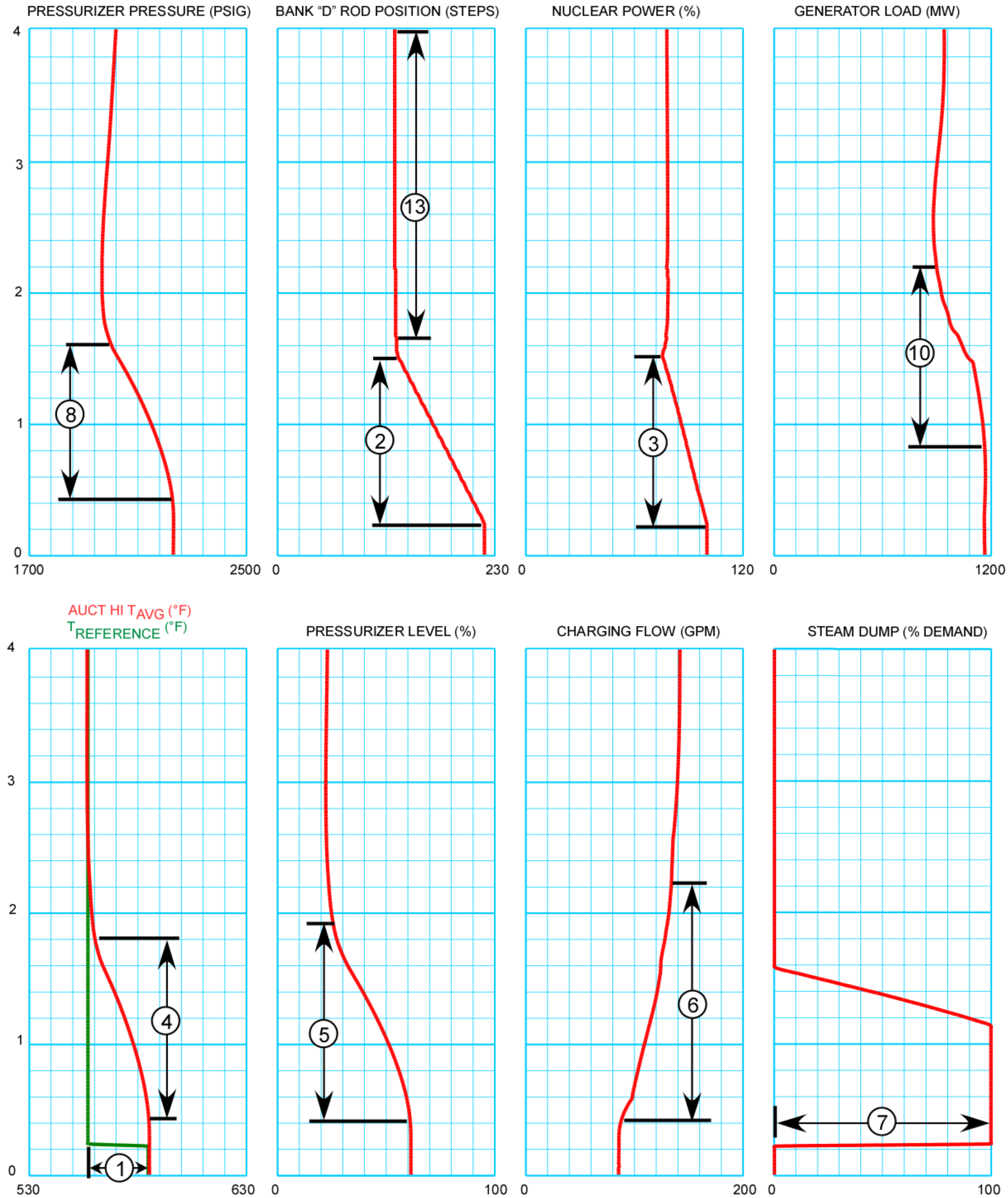
13. **Bank D rod position** stays constant over the last interval of the transient. At this point, the primary-to-secondary power mismatch has driven T_{avg} close to the minimum T_{ref} value (decreasing the temperature mismatch calling for rod insertion), the large initial input to the power mismatch circuit from the failed transmitter has died off, and nuclear power has been decreasing faster than turbine load for a fairly long period (a power mismatch calling for rod withdrawal). The total error input to the rod control system thus drops below 1°F, and rod motion stops. Also, any developing total error that would cause rod withdrawal is prevented from affecting rod position by the C-5 (auto rod withdrawal block) interlock (the input to C-5 is from the failed-low channel).

Note: The protection channel I high steam flow setpoint is exceeded immediately with the impulse pressure channel failure because the setpoint is driven to its minimum programmed value, but a high steam flow ESF actuation is avoided because the reduction in T_{avg} is not large enough to reach the low-low T_{avg} settpoint.

What this transient illustrates:

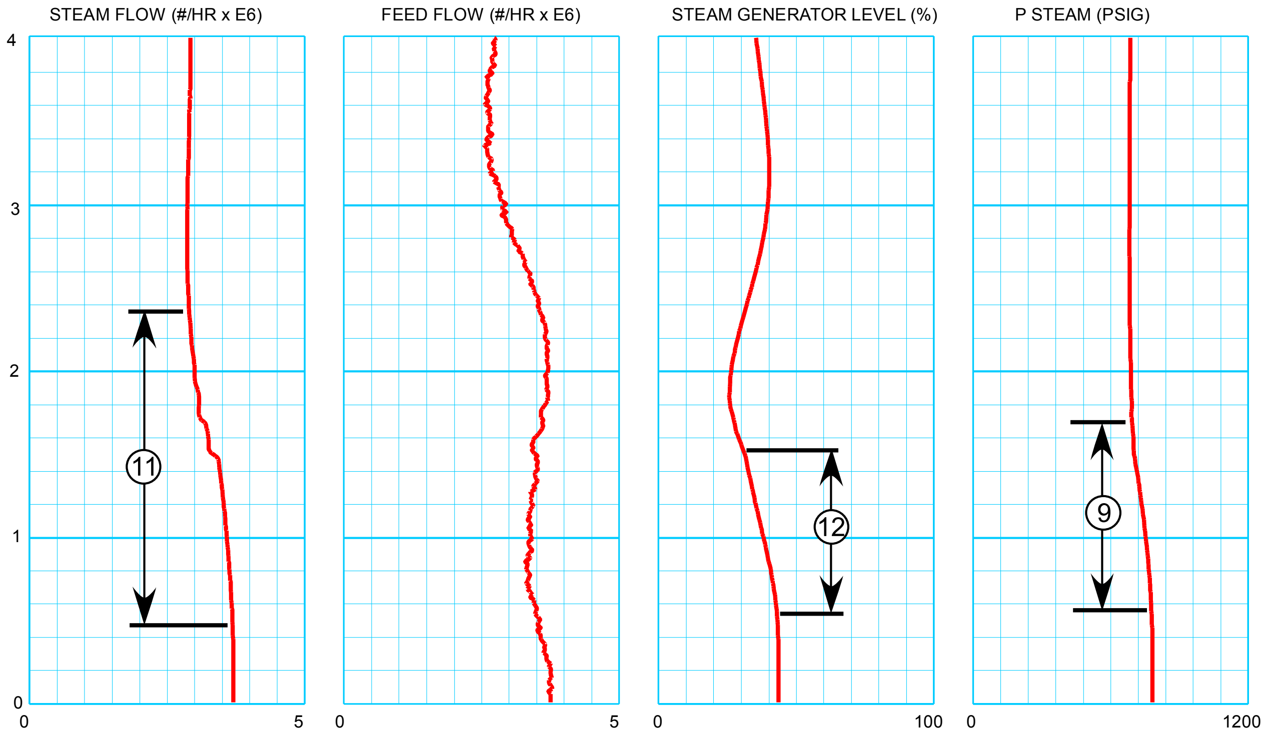
1. The responses of the rod control and steam dump control systems to a failed-low impulse pressure channel.
2. The decrease in T_{avg} when nuclear power < secondary load.

This page intentionally blank



Transient 5.35 Impulse Pressure Channel Pt-505 Fails Low

This page intentionally blank



TRANSIENT 5.35
IMPULSE PRESSURE CHANNEL
PT-505 FAILS LOW

Initial Conditions
BOL
Tavg: 585.2°F
Pressurizer Pressure: 2245 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Impulse pressure channel PT-505 fails low

Additional information:
PT-505 feeds rod control (power and temperature mismatch circuits), steam dump control (loss-of-load controller), SG water level control, and the C-5 (auto rod withdrawal block) interlock.
PT-506 feeds the C-7 (loss-of-load) interlock.
Each transmitter feeds the P-13 (turbine at-power) permissive and a high steam flow ESF setpoint generator in a separate protection channel.

Transient 5.35 Impulse Pressure Channel Pt-505 Fails Low

This page intentionally blank

Transient 5.36 Impulse Pressure Channel PT-505 Fails High

Initial Conditions:

BOL	Bank D Rod Position: 148 steps
T_{avg}: 569.5°F	Nuclear Power: 50%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Impulse pressure channel PT-505 fails high

Note: This impulse pressure channel feeds the rod control system (T_{ref} for the temperature mismatch circuit and P_{imp} for the power mismatch circuit), the steam dump control system (T_{ref} for the loss-of-load controller), and the steam generator water level control system (the level setpoint varies linearly with P_{imp} from 0 - 20% load); provides one input to the P-13 permissive (turbine at-power permissive); establishes the high steam flow setpoint for the high steam flow + low-low T_{avg} or low steam pressure ESF actuation in protection channel I; and supplies the C-5 interlock (automatic rod withdrawal block < 15% load).

Point **Explanation**

1. $\mathbf{T_{ref}}$ undergoes a rapid increase to its maximum value of 584.7°F (programmed T_{avg} for full load).
2. **Bank D rod position** increases at the maximum rate (72 steps/min) because of the large temperature mismatch ($T_{ref} >> T_{avg}$) and large power mismatch (turbine load increasing rapidly relative to nuclear power) inputs to the rod control system calling for fast rod withdrawal. The large mismatches result from the failed-high impulse pressure channel.
3. **Nuclear power** increases due to the positive reactivity associated with the rod withdrawal. The negative reactivity insertion resulting from the increase in reactor coolant temperature, as discussed in point 4 below, is not enough to counteract it.
4. $\mathbf{T_{avg}}$ increases due to the large imbalance between nuclear power and turbine load, with nuclear power increasing rapidly. As the turbine load is essentially unchanging, energy must be added to the reactor coolant, increasing its temperature.
5. The **pressurizer level** increase reflects the expansion in reactor coolant volume caused by the increase in coolant temperature.
6. **Charging flow** decreases greatly with pressurizer level high relative to the level setpoint. Both the pressurizer level and the level setpoint (a function of auctioneered high T_{avg}) are increasing, but the pressurizer level remains above the level setpoint.
7. **Pressurizer pressure** increases as the pressurizer steam bubble is squeezed by the reactor coolant expansion. Pressurizer spray and PORV operation limits the pressure increase.

Point Explanation

8. **Steam pressure** increases as heat transfer conditions in the SGs change due to the increased T_{avg}. $\dot{Q} = UA(T_{avg} - T_{stm})$; $\dot{Q}$ is essentially constant with the turbine control valve position unchanging, so T_{stm} (and P_{stm}) is gradually increasing to maintain the same ΔT across the SG tubes.

9. The **steam dump demand** increase indicates that the increase in nuclear power has driven T_{avg} higher than T_{ref} in excess of the loss of load controller deadband. Since the T_{avg} signal is lead/lag compensated, the heatup rate causes this to occur at an indicated ΔT of about 3°F (instead of 5°F). Note that at this time the dumps have not actuated, as there is no loss-of-load arming signal (no drop in turbine load).

10. **Bank D rod position** decreases as the outputs of the temperature mismatch circuit ($T_{avg} > T_{ref}$) and power mismatch circuit (the large initial input from the failed transmitter has died off, and nuclear power has been increasing faster than turbine load for a fairly long period) call for rod insertion.

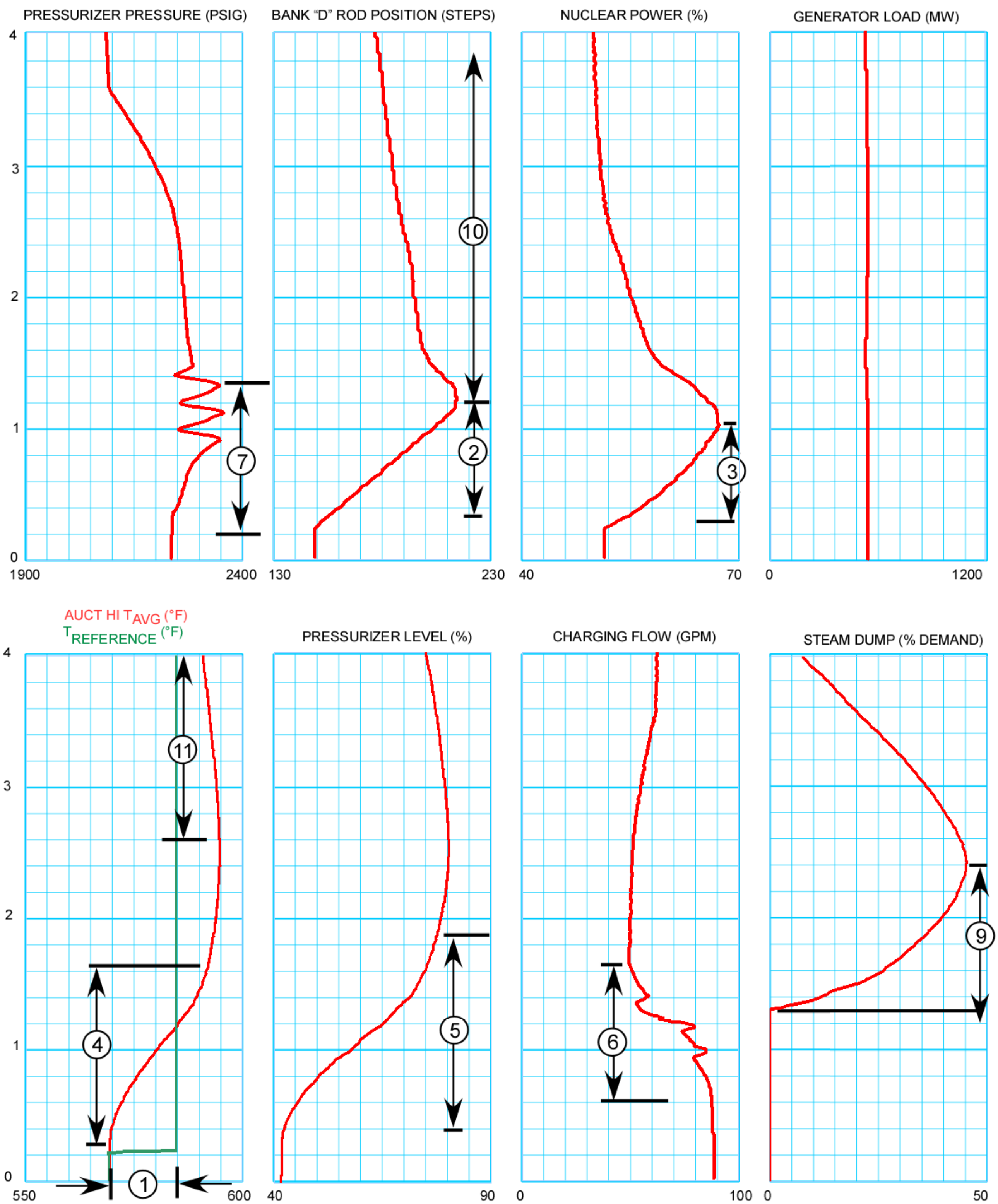
11. $\mathbf{T_{avg}}$ is decreasing as the rod insertion decreases nuclear power and narrows the primary-to-secondary power mismatch. At steady-state (not yet reached at 4 min), nuclear power will have been made equal to turbine load, and the plant will be operating at ~ 50% power with the full-load programmed T_{avg} value. The new steady state will involve both a higher bank D rod position and a higher coolant temperature, with the associated reactivity changes canceling each other.

Note 1: The starting bank D rod position of 148 steps is not typical for a plant operating at 50% power. The rods were initially diluted in to their starting positions so that the response to the failure would be more dramatic. Trojan's relaxed axial offset technical specification is not very restrictive on AFD, so starting with the bank D rods deeply inserted does not place the plant in an action statement. If this transient is started from the normal 100% conditions, the rods would be withdrawn to the 223-step auto withdrawal limit and then stop, and the slight change in reactivity would minimally affect other plant parameters, making for quite an uninteresting transient.

Note 2: The undershoot in pressurizer pressure during the second minute is caused by "overzealous" spraying. T_{avg} and pressurizer level are still increasing at this point, so steam bubble expansion cannot be contributing to the drop in pressure.

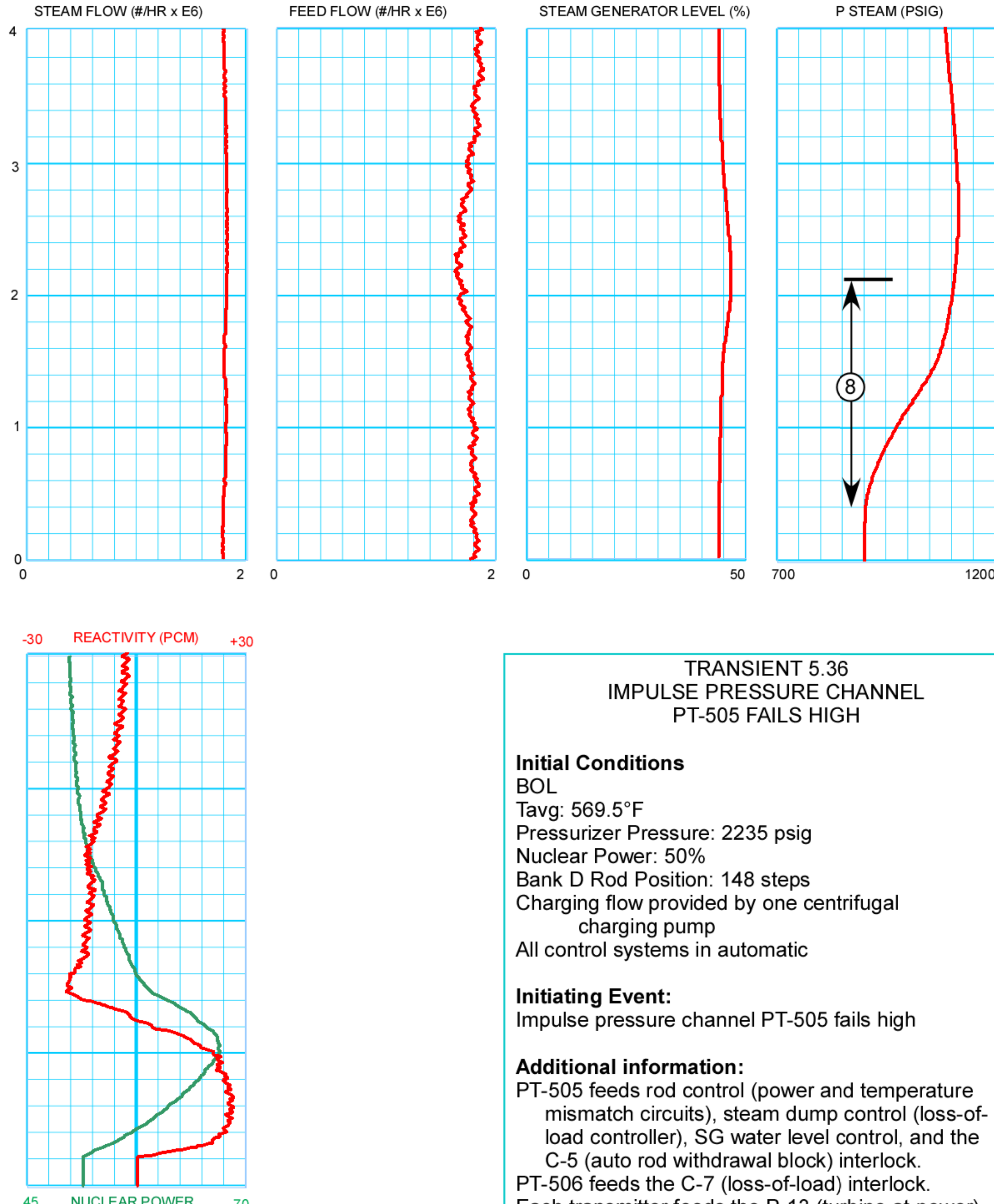
What this transient illustrates:

1. The responses of the rod control and steam dump control systems to a failed-high impulse pressure channel.
2. The increase in T_{avg} when nuclear power > secondary load.
3. The throttle pressure compensation circuit keeps generator load fairly constant over a range of steam pressures.



Transient 5.36 Impulse Pressure Channel Pt-505 Fails High

This page intentionally blank



TRANSIENT 5.36
IMPULSE PRESSURE CHANNEL
PT-505 FAILS HIGH

Initial Conditions
BOL
Tavg: 569.5°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 50%
Bank D Rod Position: 148 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Impulse pressure channel PT-505 fails high

Additional information:
PT-505 feeds rod control (power and temperature mismatch circuits), steam dump control (loss-of-load controller), SG water level control, and the C-5 (auto rod withdrawal block) interlock.
PT-506 feeds the C-7 (loss-of-load) interlock.
Each transmitter feeds the P-13 (turbine at-power) permissive and a high steam flow ESF setpoint generator in a separate protection channel.

Transient 5.36 Impulse Pressure Channel Pt-505 Fails High

This page intentionally blank

Transient 5.41 Controlling Pressurizer Pressure Channel Fails High

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 584.7°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	All control systems in automatic

Initiating Event: Controlling pressurizer pressure channel (PT-455) fails high

Point **Explanation**

1. Actual **pressurizer pressure** trends down with maximum pressurizer spray. The pressurizer pressure control system is responding to a pressure input of 2500 psig (well above the pressure setpoint of 2235 psig) from the failed channel.

2. The **pressurizer level** increase over the first ~ 2.5 min of the transient is due to an insurge as RCS water expands. Water is slightly compressible. RCS compressibility is enhanced by nucleate boiling in the core.

3. **Generator load** decreases after 2 min due to an OT∆T runback. The decreasing pressure on the three "good" channels is lowering the OT∆T trip and runback setpoints.

4. **Bank D rod position** and **nuclear power** decrease as the rod control system calls for inward rod motion in response to the turbine load decrease resulting from the runback (both the temperature mismatch and power mismatch circuits are calling for inward rod motion).

5. The reactor trips (indicated by the step drop in **bank D rod position**) on OT∆T. Because the low pressurizer pressure trip is rate sensitive, it is a possible cause of the trip, but did not initiate the trip in this transient.

6. The low **pressurizer pressure** ESF actuation setpoint is reached at ~ 2 min, 55 sec.

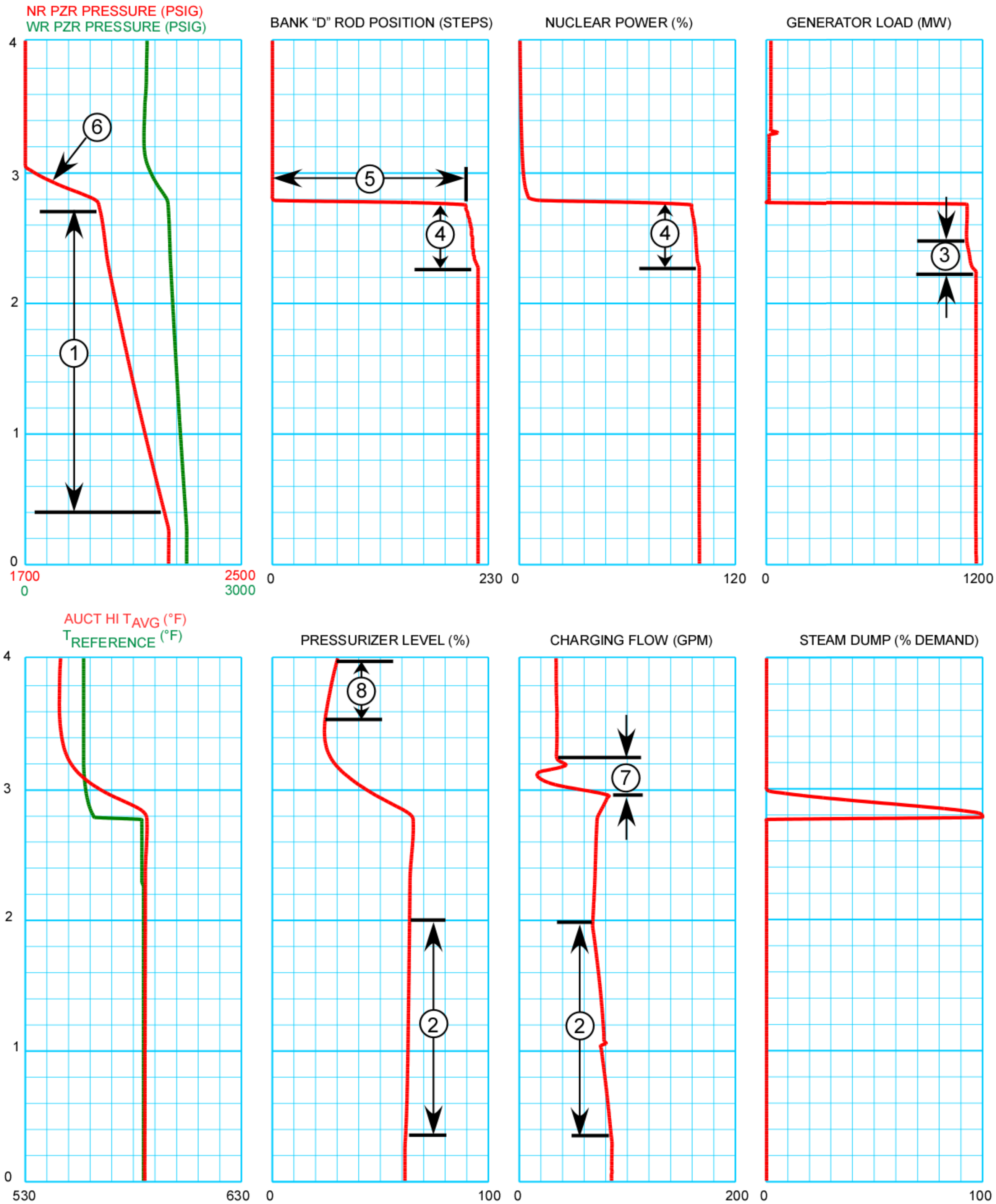
7. **Charging flow** makes its characteristic response to an ESF actuation. The post-ESF charging flow value is relatively low because the pressurizer level is near the no-load setpoint and thus charging flow control valve FCV-121 is not wide open.

8. **Pressurizer level** increases with ECCS flow and letdown isolation at the end of the transient.

What this transient illustrates:

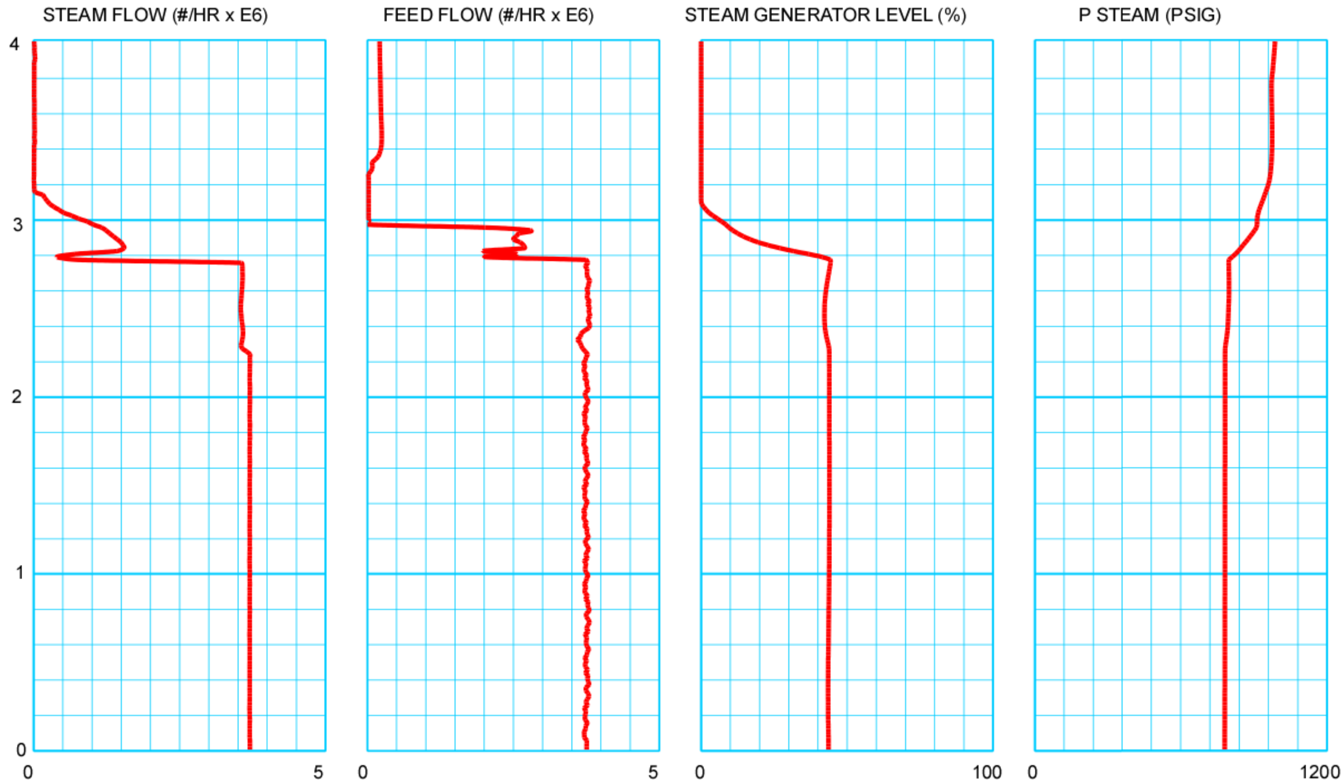
1. The response of the pressurizer pressure control system (maximum spray) to a failed-high controlling pressurizer pressure channel.
2. An increase in indicated pressurizer level with a decrease in pressurizer pressure.
3. An OT∆T reactor trip.
4. A low pressurizer pressure ESF actuation.

This page intentionally blank



Transient 5.41 Controlling Pressurizer Pressure Channel Fails High

This page intentionally blank



TRANSIENT 5.41
CONTROLLING PRESSURIZER PRESSURE
CHANNEL FAILS HIGH

Initial Conditions
BOL
Tavg: 584.7°F
Pressurizer Pressure: 2230 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Controlling pressurizer pressure channel fails high

Transient 5.41 Controlling Pressurizer Pressure Channel Fails High

This page intentionally blank

Transient 5.42 Controlling Pressurizer Level Channel Fails Low

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Controlling pressurizer level channel (LT-459) fails low

Point **Explanation**

1. **Charging flow** increases to maximum as charging flow control valve FCV-121 opens fully. The pressurizer level control system is responding to a level input of 0%, which is well below the level setpoint of 61.5%.

2. **Pressurizer level** increases rapidly with maximum charging and no letdown (one letdown isolation valve and the previously open orifice isolation valve have been closed in response to the < 17% level of the failed channel).

3. The reactor trip (as indicated by the step drop in **bank D rod position**) is caused by high pressurizer level (92%), as indicated by the 2 "good" level channels.

4. **Charging flow** increases after the trip because plant pressure drops as T_{avg} decreases to < no-load T_{avg}. The operating charging pump's output increases when the discharge pressure drops (characteristic of centrifugal pump operation).

5. **Pressurizer level** recovers after the trip due to (a) continued charging (at a high rate) and plant heatup and (b) continued charging alone after the plant heatup is stopped at no-load T_{avg} by steam dump operation (the end of the post-trip heatup accounts for the slope change in the pressurizer level recovery). Letdown is still isolated. The pressurizer eventually fills.

6. **VCT level** drops during the interval of 0 - 19 min because maximum charging and isolated letdown are depleting the VCT inventory. The change in slope at ~ 3 min indicates initiation of automatic makeup, which slows but does not stop the VCT level decrease.

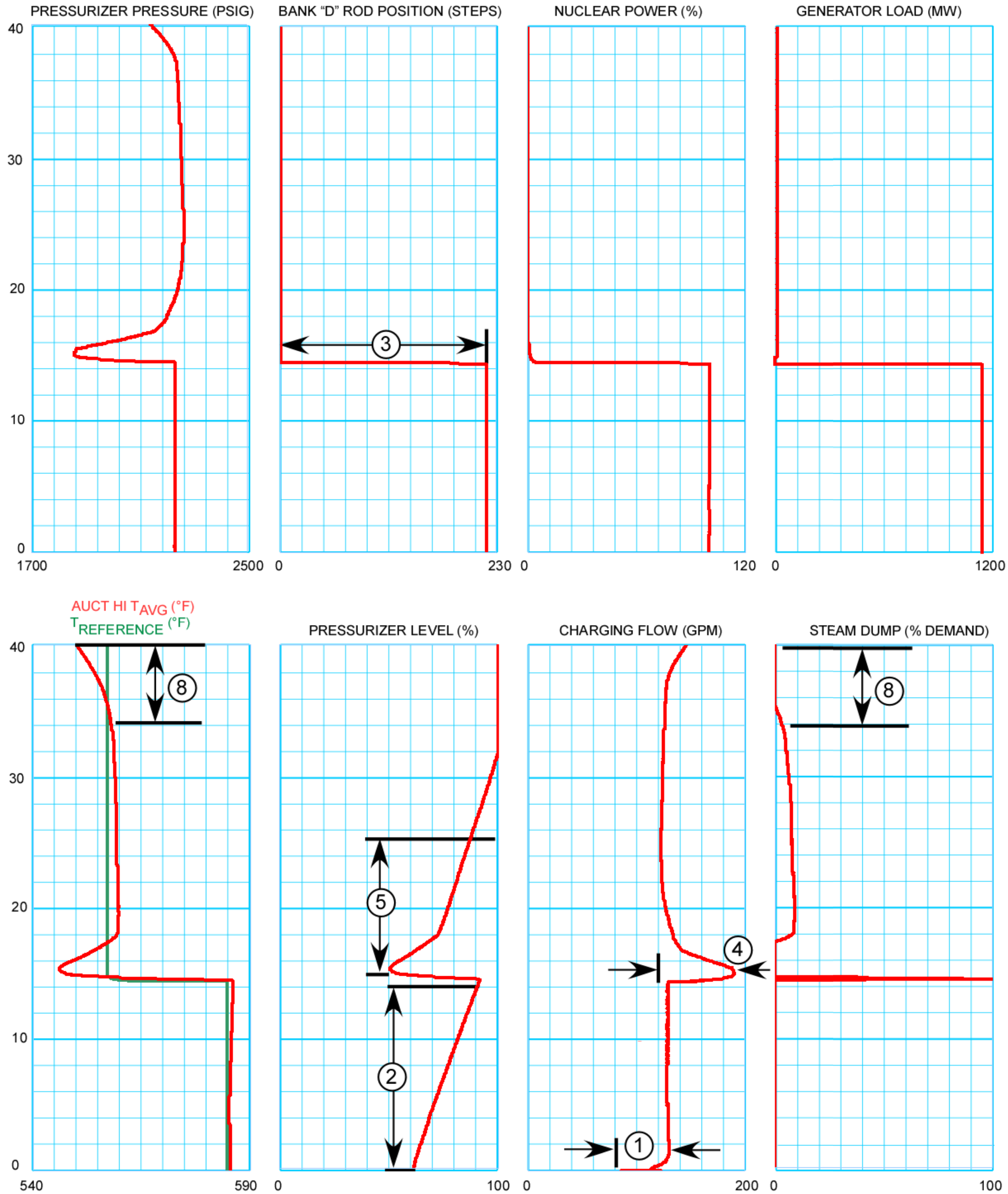
7. **VCT level** recovers after 19 min with continued automatic makeup because the VCT outlet valves have shut on low VCT level (the charging pump suctions are now supplied by the RWST). Recovery of the VCT ends with the cutoff of automatic makeup.

<u>Point</u> <u>Explanation</u>

8. RCS temperature drops when decay heat falls below the level of heat being removed by auxiliary feedwater (AFW) flow. It is necessary to throttle AFW flow to control RCS temperature at this point. One symptom that AFW flow needs to be throttled is closure of all steam dump valves. If AFW is not throttled, the cooldown will result in a safety injection actuation on low pressurizer pressure.

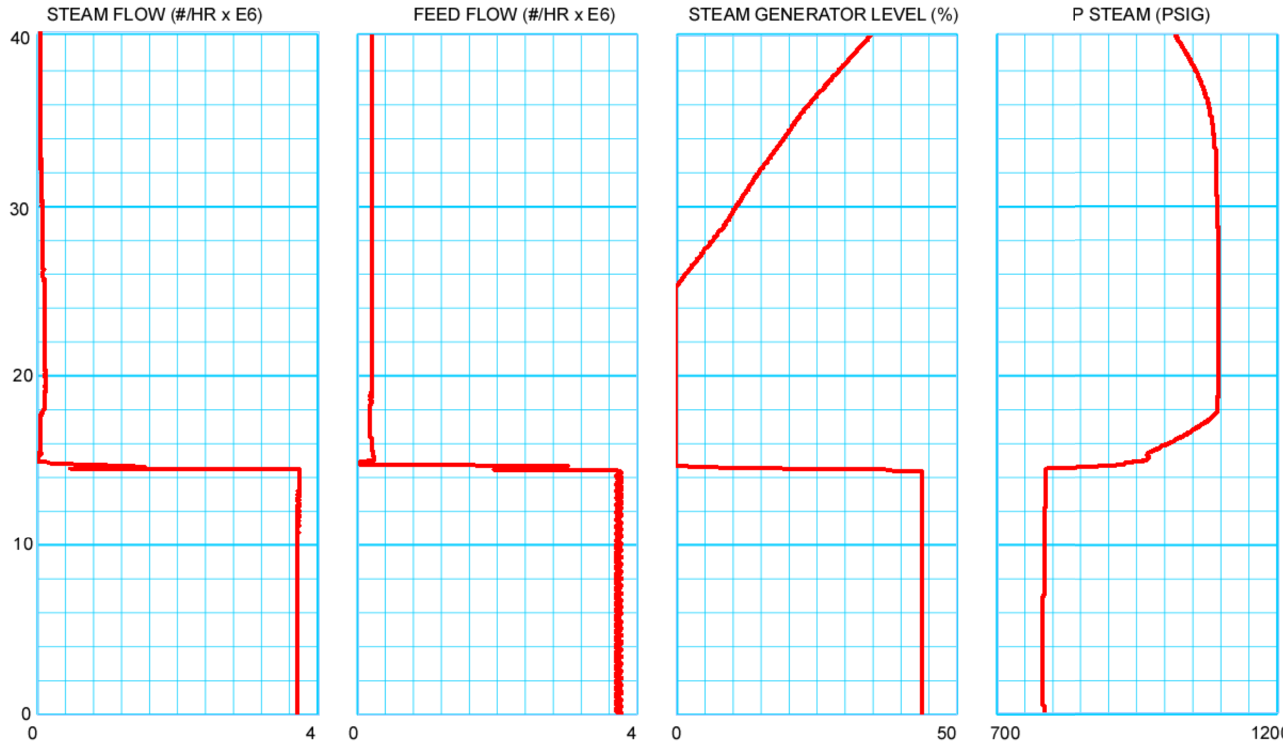
What this transient illustrates:

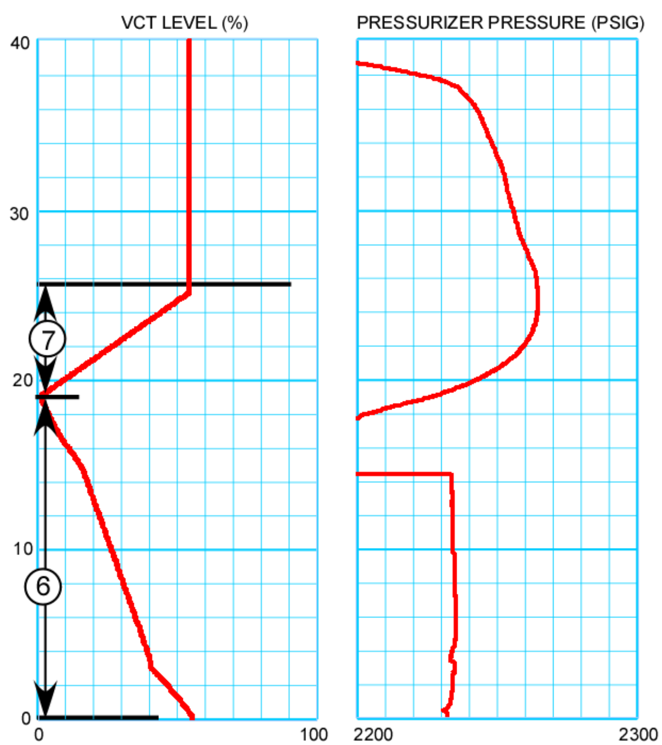
1. The response of the pressurizer level control system (maximum charging) to a failed-low controlling pressurizer level channel.
2. A high pressurizer level reactor trip.
3. The depletion and recovery of VCT level.
4. An eventual RCS post-trip cooldown with maximum AFW flow.



Transient 5.42 Controlling Pressurizer Level Channel Fails Low

This page intentionally blank





TRANSIENT 5.42
CONTROLLING PRESSURIZER LEVEL
CHANNEL FAILS LOW

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
Seal return flow is supplied to the suction of the CCPs.
All control systems in automatic

Initiating Event:
Controlling pressurizer level fails low

Transient 5.42 Controlling Pressurizer Level Channel Fails Low

This page intentionally blank

Transient 5.43 Controlling Pressurizer Pressure Channel Fails Low

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Controlling pressurizer pressure channel (PT-455) fails low

Point **Explanation**

1. Actual **pressurizer pressure** trends up with maximum heater output from both variable and backup heaters. The pressurizer pressure control system is responding to a pressure input of 1700 psig (well below the pressure setpoint of 2235 psig) from the failed channel.

2. On five different occasions, the **pressurizer pressure** increases to the PORV lift setpoint. Only PORV PCV-456 lifts; the 2-out-of-2 coincidence for PCV-455A cannot be met with one of its inputs (pressure channel 455) failed low. With each PORV lift, pressure drops about 40 psig, the PORV recloses in a few seconds, and the upward trend in pressure resumes with maximum heater output.

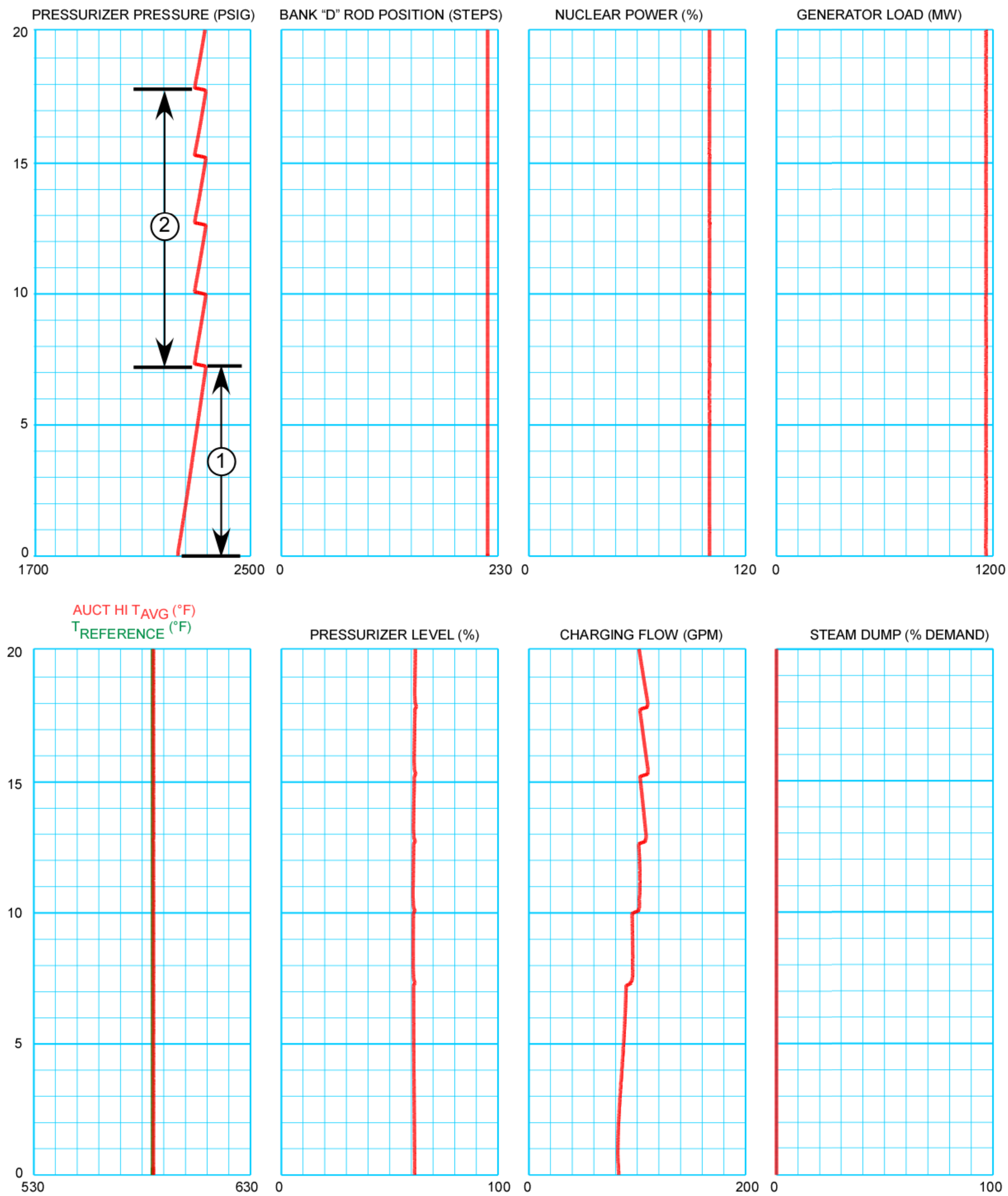
3. The decrease in **pressurizer level** reflects a pressurizer outsurge. Water is slightly compressible. RCS compressibility is enhanced by nucleate boiling in the core.

4. The five short-term increases (“blips”) in **pressurizer level** are coincident with the PORV lifts. Each PORV lift results in a decrease in reactor coolant density and corresponding increase in pressurizer level.

What this transient illustrates:

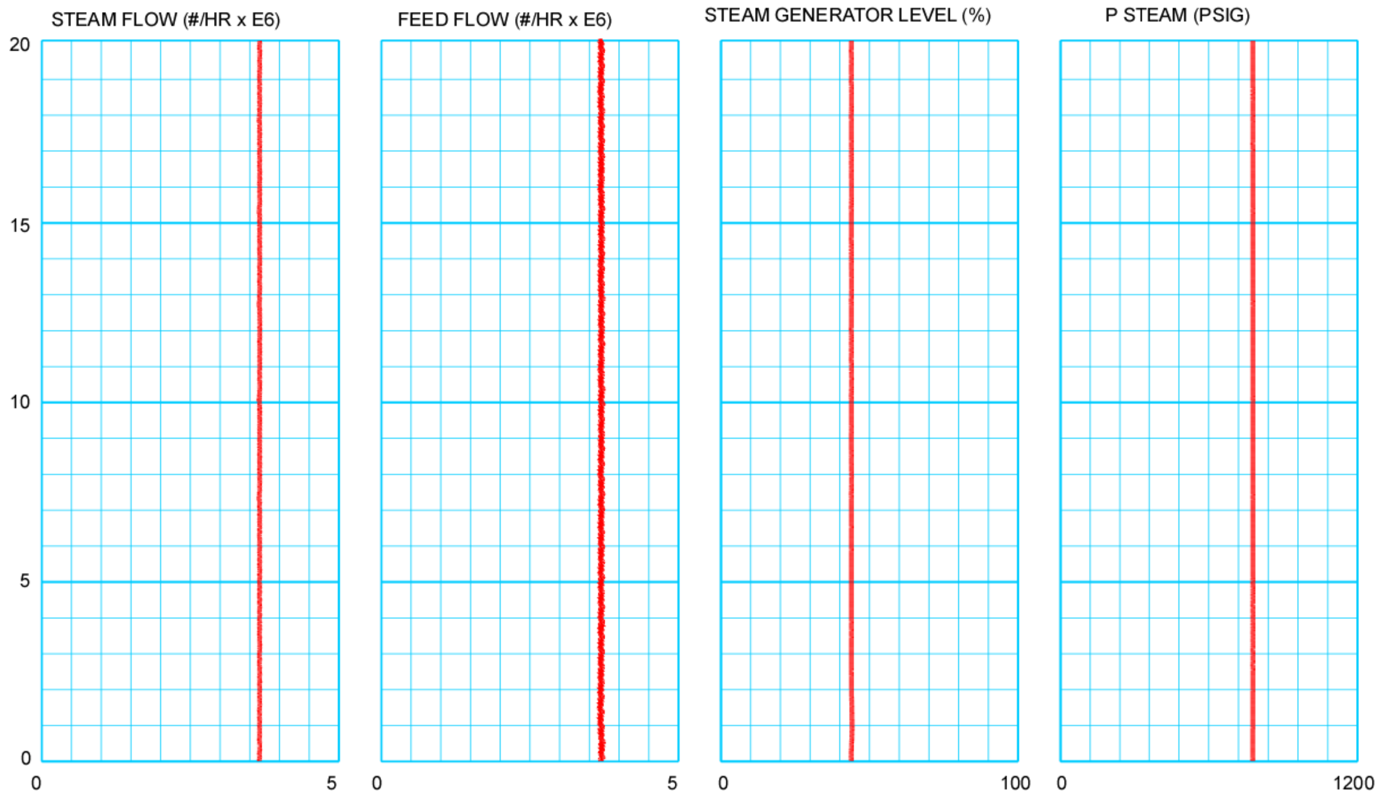
1. The response of the pressurizer pressure control system (maximum heaters) to a failed-low controlling pressurizer pressure channel.
2. The responses of pressurizer pressure, pressurizer level, and charging flow to several PORV lifts.

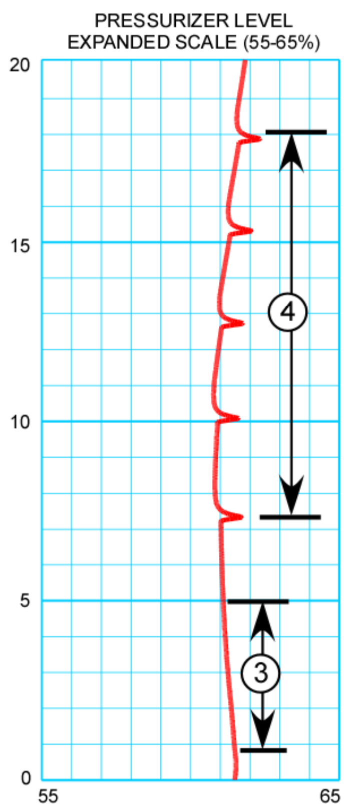
This page intentionally blank



Transient 5.43 Controlling Pressurizer Pressure Channel Fails Low

This page intentionally blank





TRANSIENT 5.43
CONTROLLING PRESSURIZER PRESSURE
CHANNEL FAILS LOW

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Controlling pressurizer pressure fails low

Additional Information:
Pressurizer level expanded scale

Transient 5.43 Controlling Pressurizer Pressure Channel Fails Low

This page intentionally blank

Transient 5.51 Controlling Steam Generator Level Channel Fails Low

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Controlling SG #1 level channel (LT-519) fails low

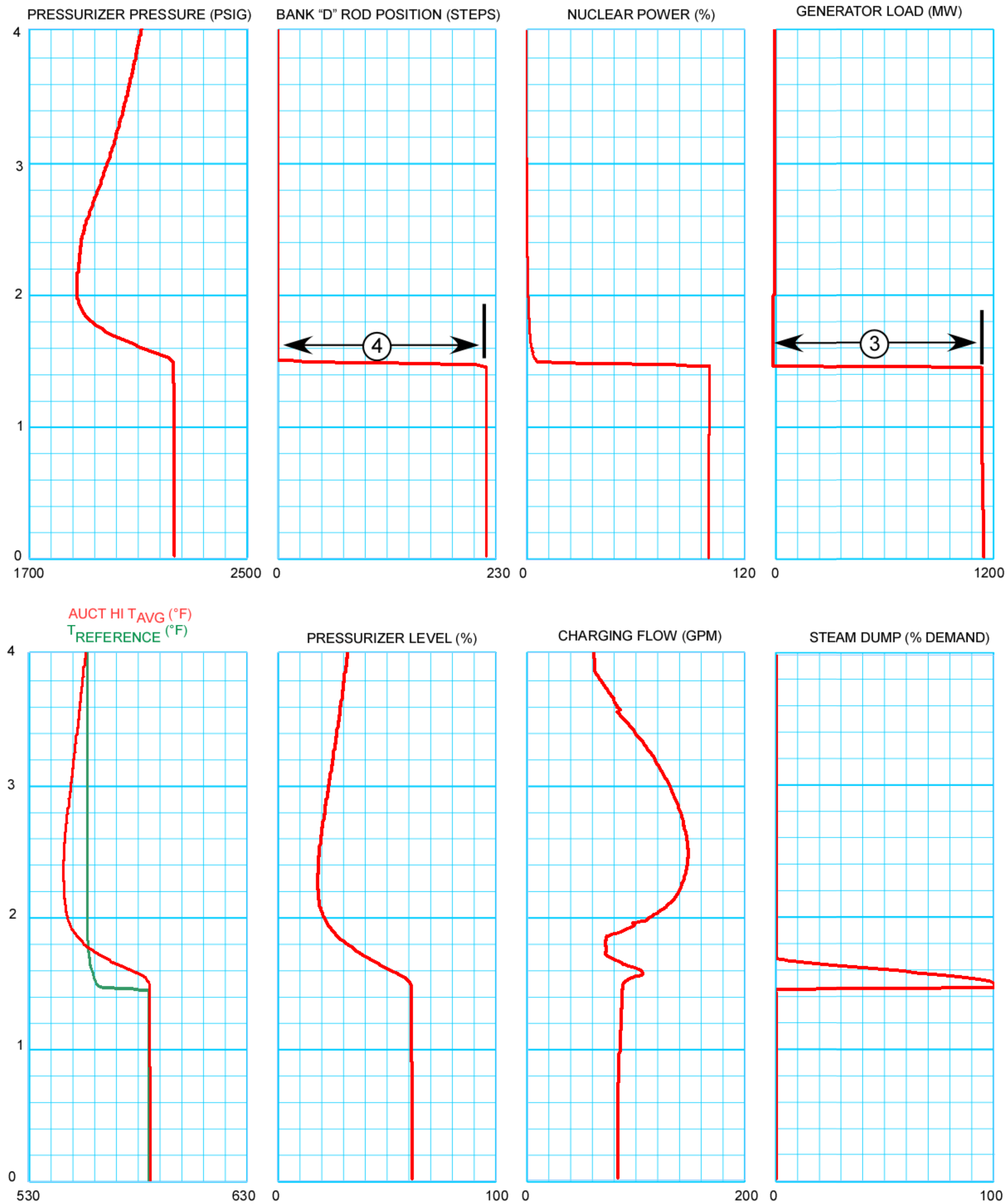
Point Explanation

1. **Feed flow to SG #1** increases to maximum as the main feed regulating valve opens in response to the controlling level input falling well below the 44% level setpoint.
2. **Steam generator level in SG #1** increases rapidly with the feed flow increase and no change in steam flow.
3. The turbine trip (as indicated by the step drop in **generator load**) is caused by high steam generator water level (the level measured by the two “good” level detectors on SG #1 reaches the 69% level turbine trip setpoint).
4. The reactor trip (as indicated by the step drop in **bank D rod position**) is caused by the turbine trip with plant power above the P-7 setpoint.

What this transient illustrates:

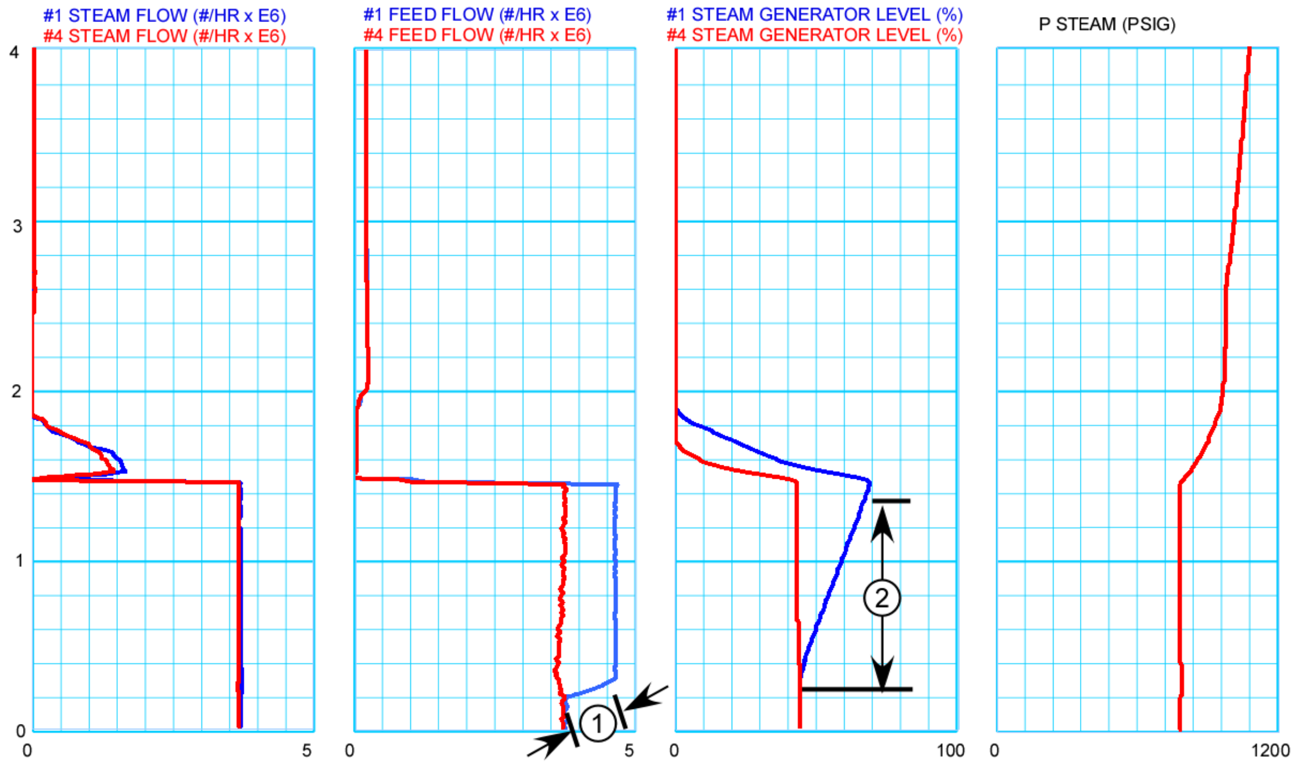
1. The response of the steam generator water level control system (maximum feed regulating valve position) to a failed-low controlling steam generator level channel.
2. A turbine trip on high SG level.
3. A reactor trip on a turbine trip + P-7.

This page intentionally blank



Transient 5.51 Controlling Steam Generator Level Channel Fails Low

This page intentionally blank



TRANSIENT 5.51
STEAM GENERATOR CONTROLLING
LEVEL CHANNEL FAILS LOW

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Steam Generator #1 controlling level transmitter fails low

Transient 5.51 Controlling Steam Generator Level Channel Fails Low

This page intentionally blank

Transient 5.52 Controlling Steam Generator Level Channel Fails High

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Controlling SG #1 level channel (LT-519) fails high

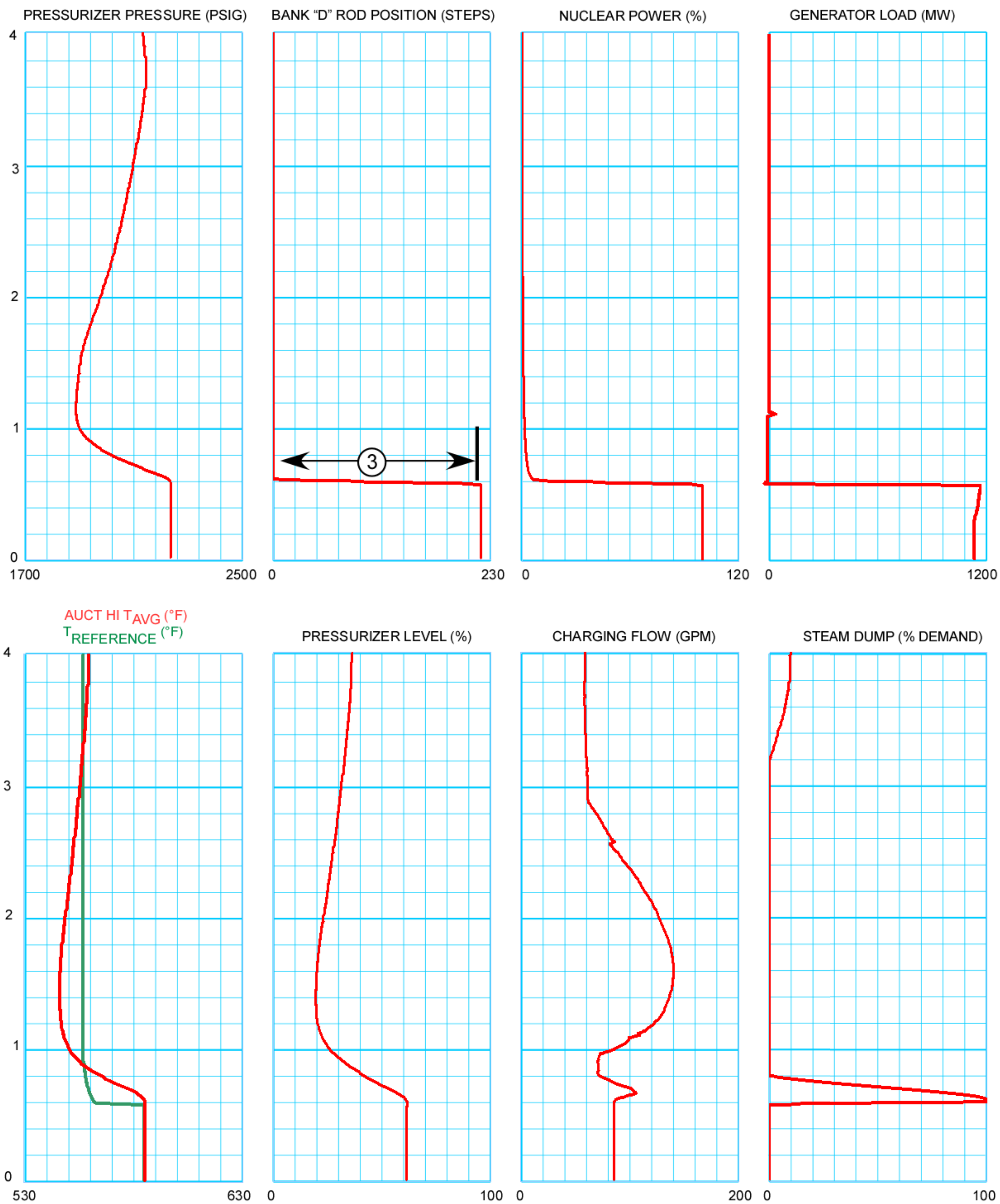
Point **Explanation**

1. **Feed flow to SG #1** decreases to minimum as the main feed regulating valve closes in response to the controlling level input rising well above the 44% level setpoint.
2. **Steam generator level in SG #1** decreases rapidly with the feed flow decrease and no change in steam flow.
3. The reactor trips (as indicated by the step drop in **bank D rod position**) on low steam generator water level (25.5%, as measured by the two "good" SG #1 level channels) and steam flow/feed flow mismatch.

What this transient illustrates:

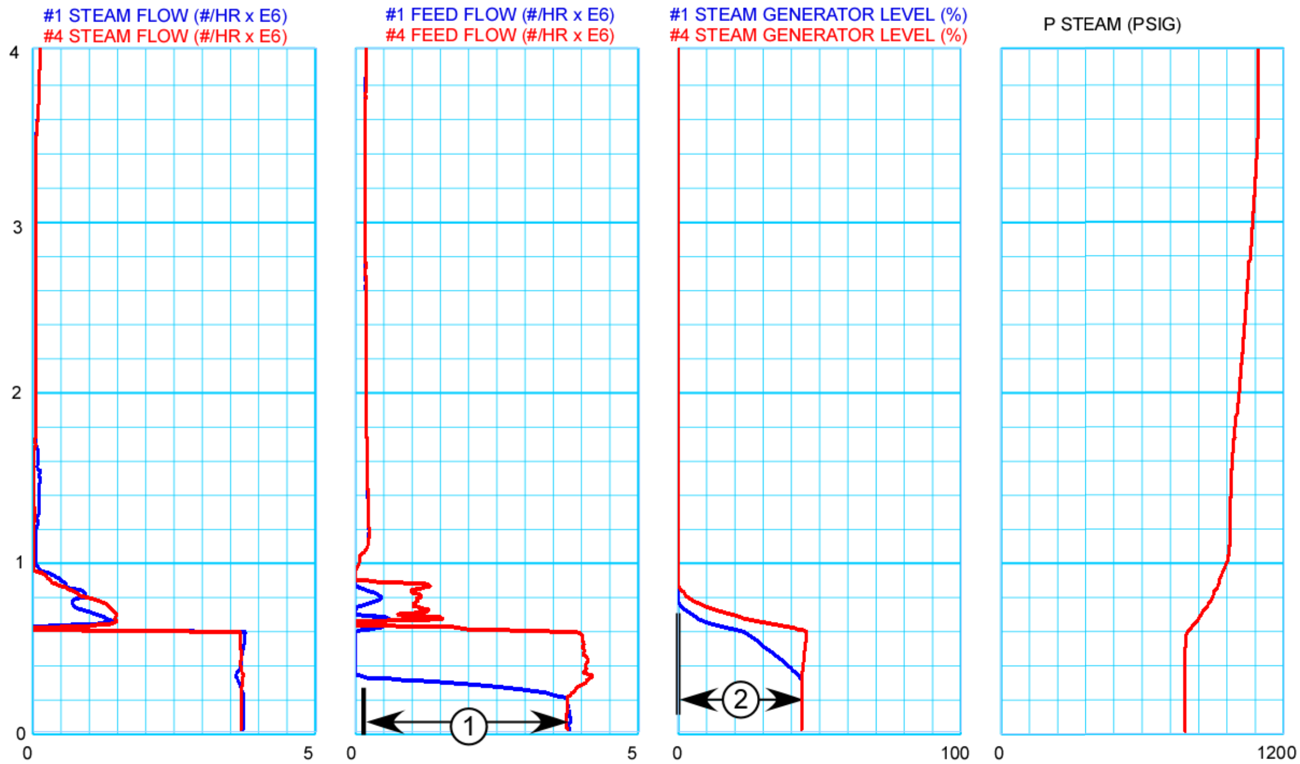
1. The response of the steam generator water level control system (minimum feed regulating valve position) to a failed-high controlling steam generator level channel.
2. A reactor trip on low SG level + steam flow/feed flow mismatch.

This page intentionally blank



Transient 5.52 Controlling Steam Generator Level Channel Fails High

This page intentionally blank



TRANSIENT 5.52
STEAM GENERATOR CONTROLLING
LEVEL CHANNEL FAILS HIGH

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Steam Generator #1 controlling level transmitter fails high

Transient 5.52 Controlling Steam Generator Level Channel Fails High

This page intentionally blank

Transient 5.53 Controlling Steam Generator Feed Flow Channel Fails Low

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Controlling SG #1 feed flow channel (FT-510) fails low

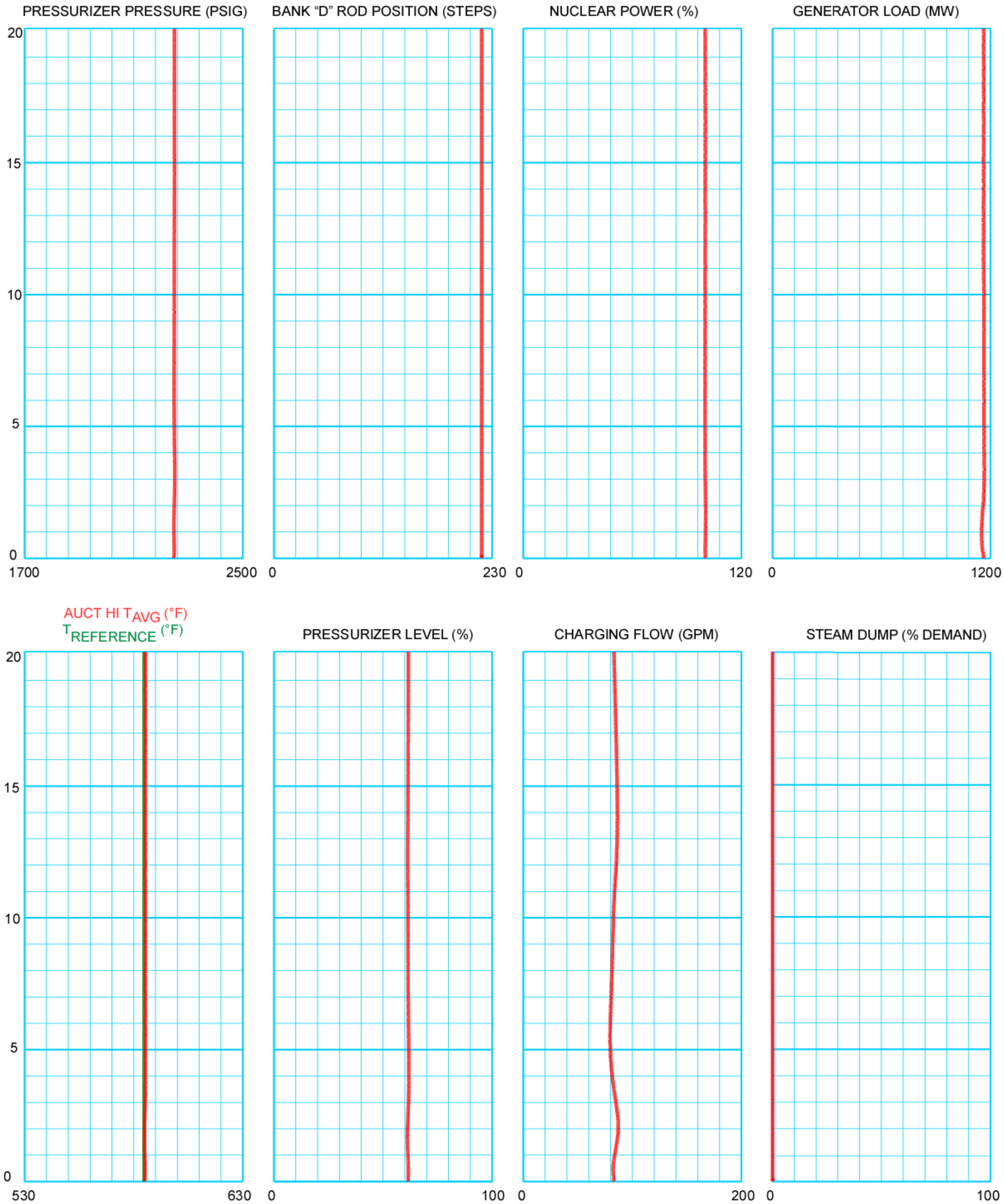
Point **Explanation**

1. **Feed flow to SG #1** increases to maximum as the main feed regulating valve opens in response to the large steam flow/feed flow mismatch (feed flow < steam flow) input to the steam generator water level control system.
2. **Steam generator level in SG #1** increases with the feed flow increase and no change in steam flow. Even after feed flow begins to decrease (as explained in point 3 below), steam generator level continues to rise as long as feed flow exceeds steam flow.
3. The large (integrated) level error causes **feed flow to SG #1** to decrease, overriding the still-present steam flow/feed flow mismatch. This trend in feed flow illustrates that steam generator water level control is level dominant.
4. **Steam generator level in SG #1** decreases with feed flow less than steam flow. The level error input to steam generator water level control brings level back to setpoint (44%).
5. **Steam generator level in SG #1** remains constant at setpoint over the final 10 min of the transient. The integrated level error from the first 10 min of the transient (level > setpoint) counteracts the steam flow/feed flow mismatch (controlling feed flow channel still failed low) during this interval.

What this transient illustrates:

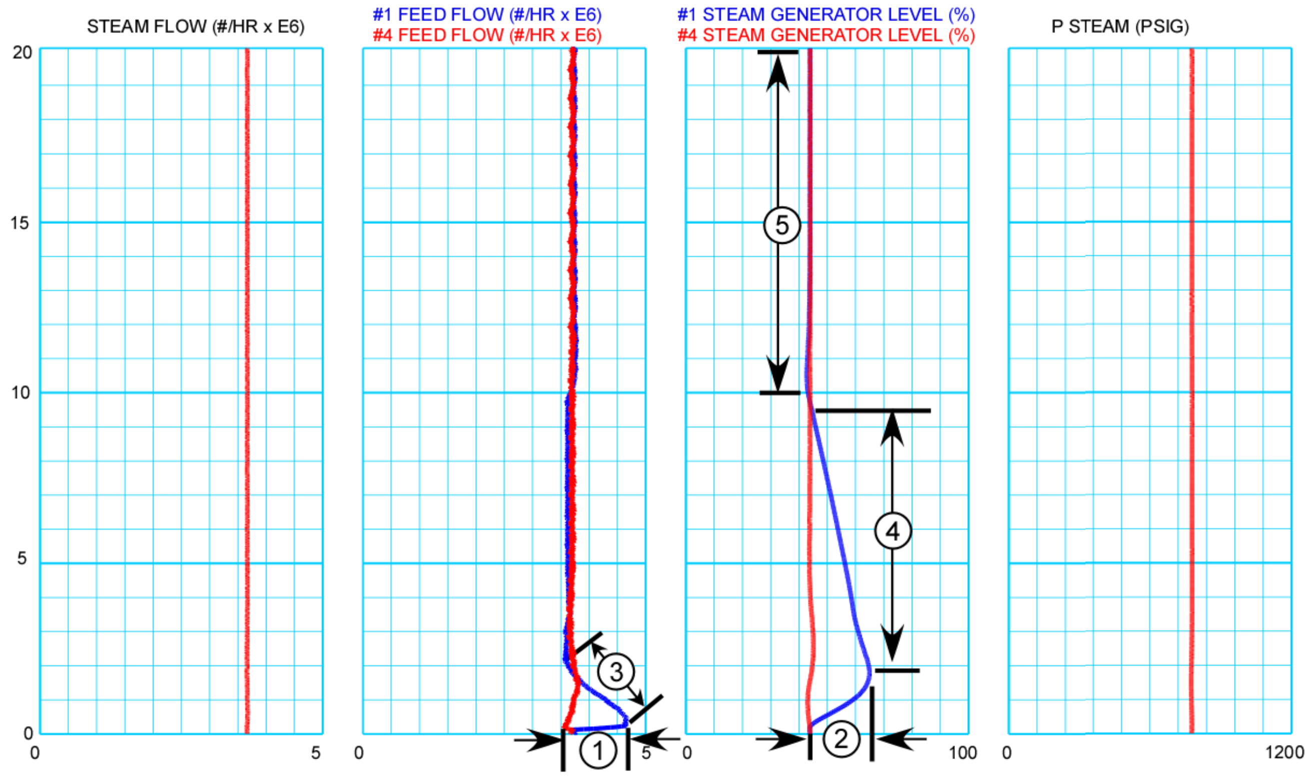
1. The response of the steam generator water level control system to a failed-low controlling steam generator feed flow channel.
2. That steam generator water level control is a level-dominant system.

This page intentionally blank



Transient 5.53 Controlling Steam Generator Feed Flow Channel Fails Low

This page intentionally blank



TRANSIENT 5.53
STEAM GENERATOR CONTROLLING
FEED FLOW CHANNEL FAILS LOW

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Steam Generator #1 controlling feed flow transmitter fails low

Transient 5.53 Controlling Steam Generator Feed Flow Channel Fails Low

This page intentionally blank

Transient 5.54 Controlling Steam Generator Feed Flow Channel Fails High

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: Controlling SG #1 feed flow channel (FT-510) fails high

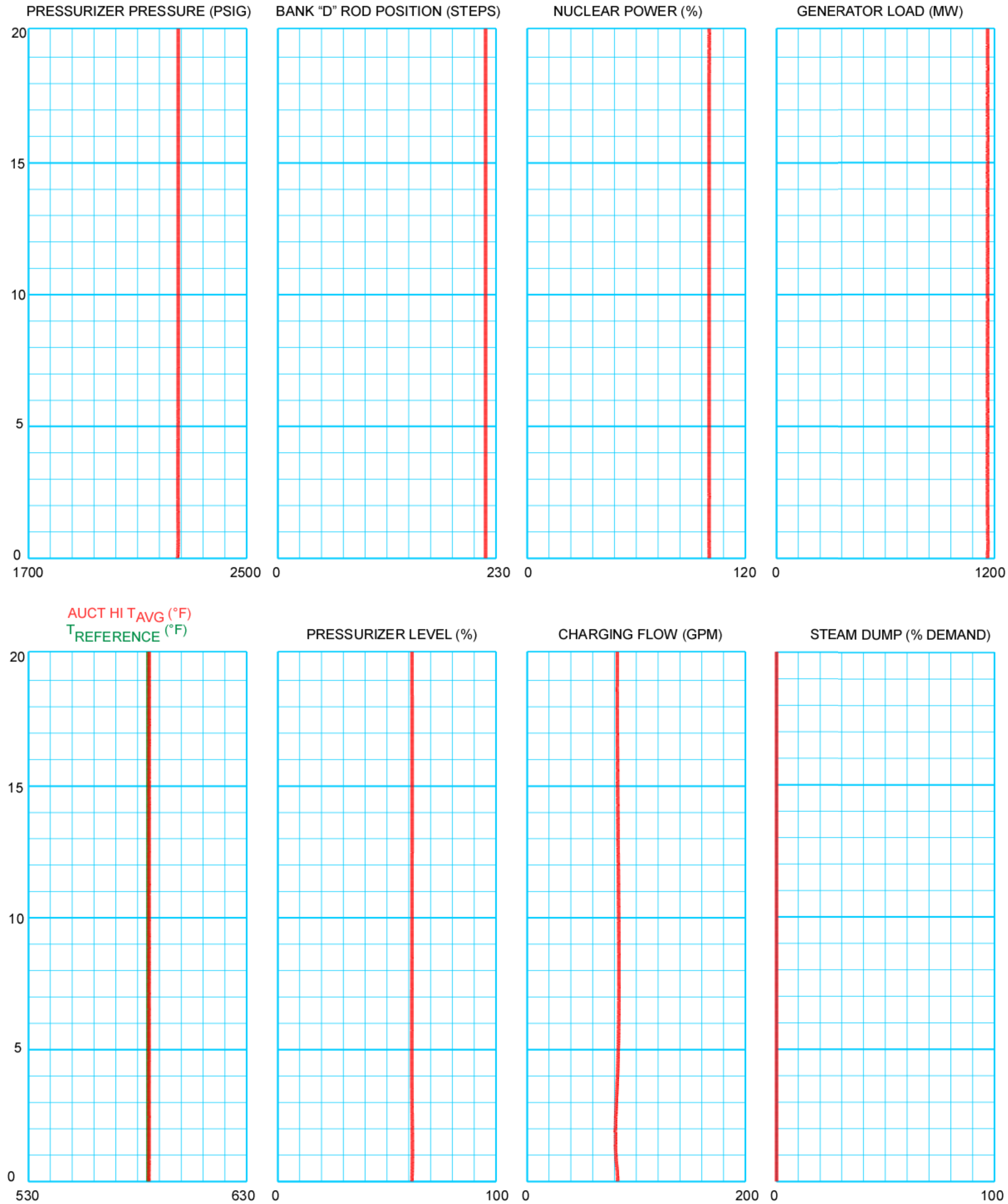
Point Explanation

1. **Feed flow to SG #1** decreases as the main feed regulating valve closes in response to the large steam flow/feed flow mismatch (feed flow > steam flow) input to the steam generator water level control system.
2. **Steam generator level in SG #1** decreases with the feed flow decrease and no change in steam flow. The change in SG level in this transient is not as great as that seen in transient 5.53, as the flow error input to the water level control system is not as large. (Feed flow failing high goes from 3.7 X 10^6 lbm/hr to 5 X10^6 lbm/hr; feed flow failing low goes from 3.7 X 10^6 lbm/hr to 0.) Even after feed flow begins to increase (as explained in point 3 below), steam generator level continues to drop as long as steam flow exceeds feed flow.
3. The (integrated) level error causes **feed flow to SG #1** to increase, overriding the still-present steam flow/feed flow mismatch. This trend in feed flow illustrates that steam generator water level control is level dominant.
4. **Steam generator level in SG #1** increases with feed flow greater than steam flow. The level error input to steam generator water level control brings level back to setpoint (44%).
5. **Steam generator level in SG #1** remains constant at setpoint over the final few minutes of the transient. The integrated level error from the first several minutes of the transient (level < setpoint) counteracts the steam flow/feed flow mismatch (controlling feed flow channel still failed high) during this interval.

What this transient illustrates:

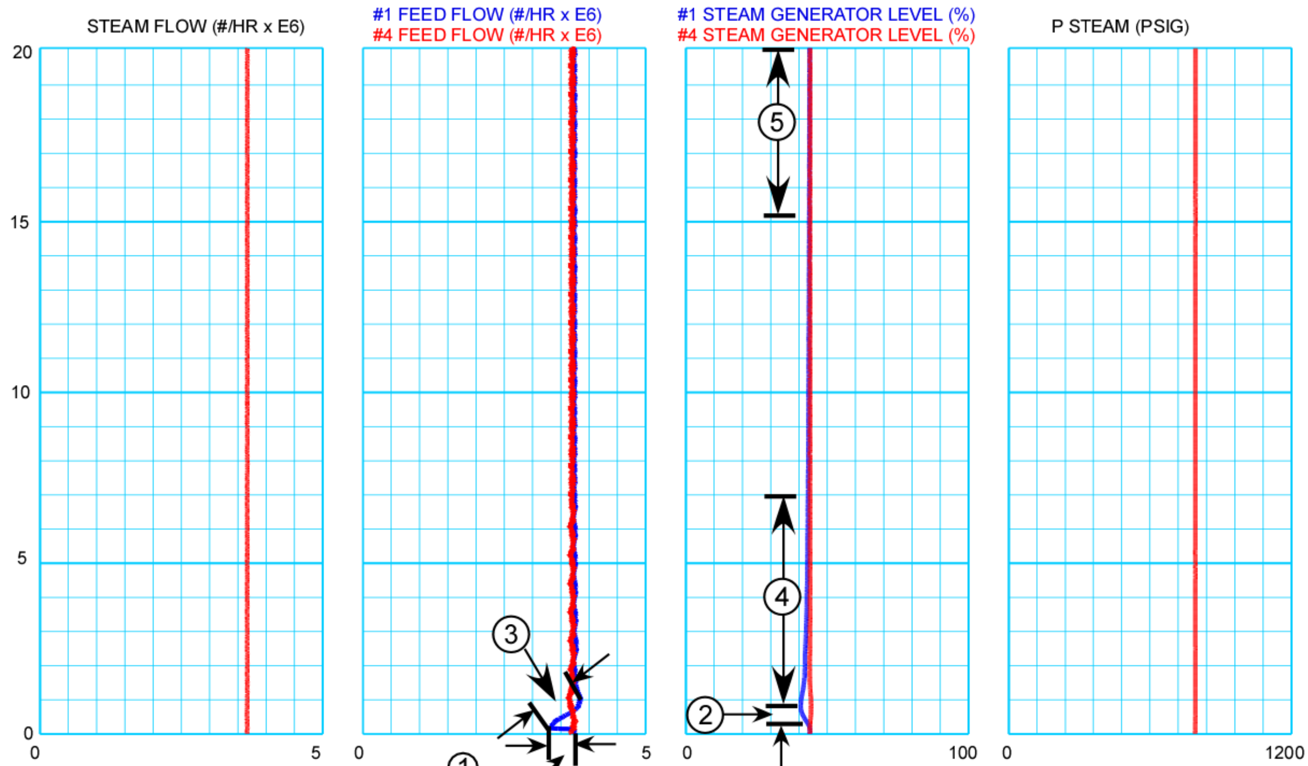
1. The response of the steam generator water level control system to a failed-high controlling steam generator feed flow channel.
2. That steam generator water level control is a level-dominant system.

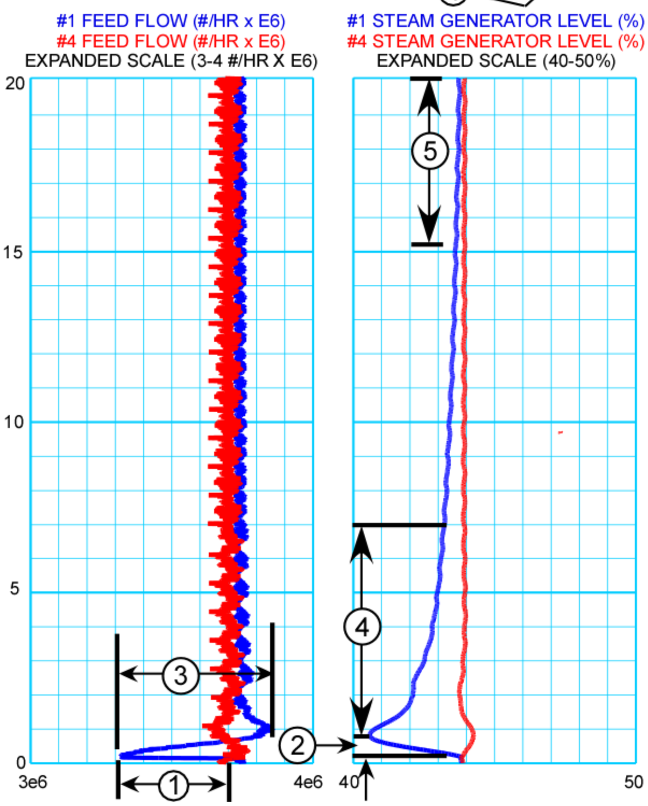
This page intentionally blank



Transient 5.54 Controlling Steam Generator Feed Flow Channel Fails High

This page intentionally blank





TRANSIENT 5.54
STEAM GENERATOR CONTROLLING
FEED FLOW CHANNEL FAILS HIGH

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Steam Generator #1 controlling feed flow transmitter fails high

Additional Information:
Feed Flow Expanded Scale
Steam Generator Level Expanded Scale

Transient 5.54 Controlling Steam Generator Feed Flow Channel Fails High

This page intentionally blank

Transient 5.61 Trip Of #1 Main Feed Pump

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.1°F	Nuclear Power: 100%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: #1 main feed pump trips

Point **Explanation**

1. **#1 MFP flow rate** decreases to 0 with the trip of the pump.

2. **#2 MFP flow rate** increases to compensate for the tripped pump. With the loss of discharge pressure from the #1 pump, the ΔP across the main feed regulating valves decreases, and the feed pump speed control system increases the speed of the still operating pump to boost feedwater pressure and regulating valve ΔP.

3. **Generator load** decreases rapidly to less than 60% on the loss-of-feed-pump turbine setback. The setback feature of the turbine EHC system reduces load at 2%/sec by rapidly reducing the input to the load limit circuit.

4. **Bank D rod position** decreases at the maximum rate (72 steps/min) in response to the power mismatch circuit (turbine load decreasing rapidly) and temperature mismatch circuit ($T_{ref} < T_{avg}$) of the rod control system.

5. **Nuclear power** decreases in response to the negative reactivity added by rod insertion and, to a small extent, by the increase in T_{avg}.

6. **Steam dump demand** increases to a large value with the drop in T_{ref} accompanying the turbine setback and the increase in T_{avg} resulting from the power mismatch (nuclear power > secondary load).

7. **Steam flow** decreases in response to the closure of the turbine control valves resulting from the setback, then increases with steam dump actuation. The steam dumps are armed by the setback-induced rapid load reduction, and a large demand exists, as described in point 6 above. The steam dump demand and arming signal exist for several seconds prior to the increase in steam flow at ~ 36 sec; before this time steam dump operation is masked by the rapid control valve closure.

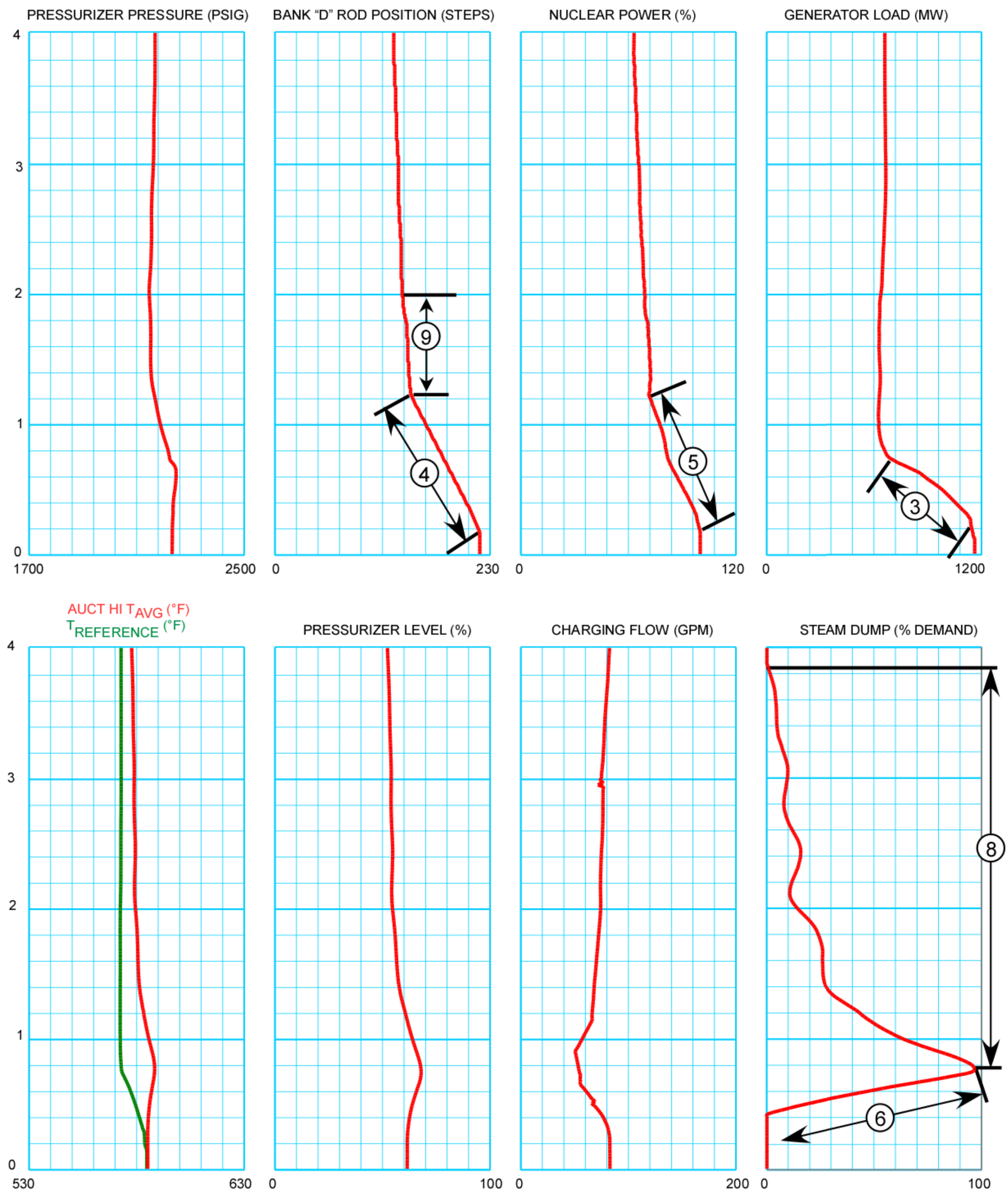
8. **Steam dump demand** decreases, first rapidly and then steadily, as rod insertion brings T_{avg} down to T_{ref}.

<u>**Point**</u> <u>**Explanation**</u>

9. The change in **bank D rod position** slows as the inputs to the rod control system from the temperature and power mismatch circuits almost cancel. T_{avg} is > T_{ref}, so the temperature mismatch circuit is calling for rod insertion, but nuclear power has been decreasing with turbine load constant, so the power mismatch circuit is calling for rod withdrawal.

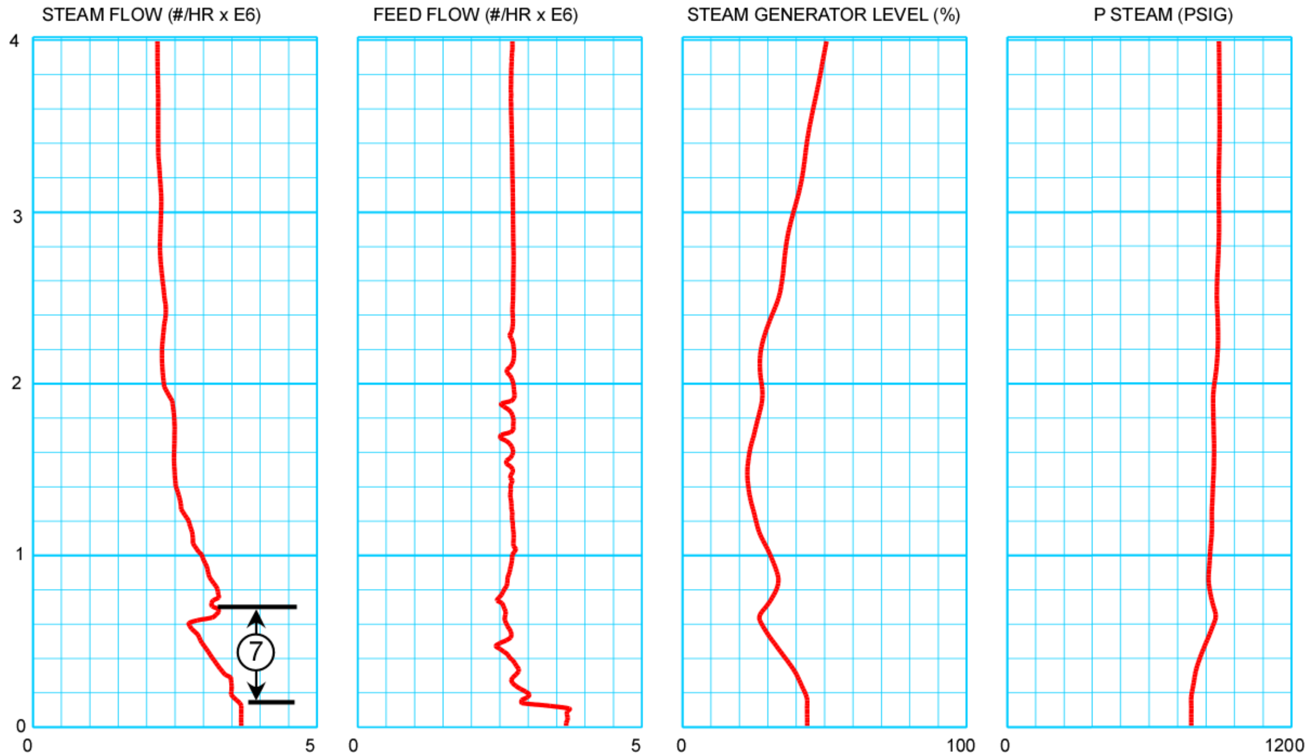
What this transient illustrates:

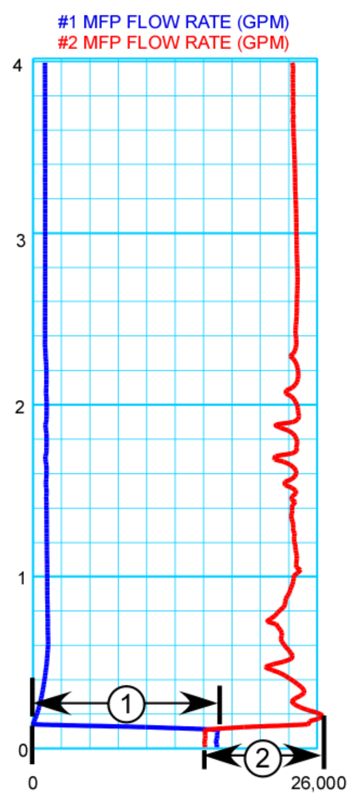
1. A loss-of-feed-pump turbine setback.
2. Steam dump actuation to handle the difference between nuclear power and secondary load.



Transient 5.61 Trip Of #1 Main Feed Pump

This page intentionally blank





TRANSIENT 5.61
TRIP OF #1 MAIN FEED PUMP

Initial Conditions
BOL
Tavg: 585.1°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
#1 main feed pump trips

Transient 5.61 Trip Of #1 Main Feed Pump

This page intentionally blank

Transient 5.62 Inadvertent MSIV Closure

Initial Conditions:

BOL	Bank D Rod Position: 194 steps
T_{avg}: 567°F	Nuclear Power: 50%
PZR Pressure: 2235 psig	All control systems in automatic

Initiating Event: The MSIV in the main steam line from the #1 SG inadvertently shut

Point **Explanation**

1. **Steam flow from the #1 SG** rapidly decreases to near 0 as the MSIV shuts. The MSIV closes in about 2 sec.

2. **Steam generator level in the #1 SG** shrinks rapidly in response to the large reduction in steam demand from that SG with the MSIV closure.

3. **Feed flow to the #1 SG** rapidly decreases as the #1 SG water level control system responds to the rapid drop in steam flow by closing the #1 SG main feedwater regulating valve.

4. **Steam pressure in the #1 main steam line** increases rapidly, as steam is bottled up in the #1 SG and its associated main steam line.

5. The removal of flow resistance associated with the isolation of the #1 main steam line causes **steam flow from the #4 SG** (as well as from the #2 and #3 SGs) to increase.

6. **Steam generator level in the #4 SG** (as well as in the #2 and #3 SGs) swells in response to the increase in steam demand from that SG.

7. **Feed flow to the #4 SG** (as well as to the #2 and #3 SGs) increases as the #4 SG water level control system responds to the increase in steam flow by opening the #4 SG main feedwater regulating valve.

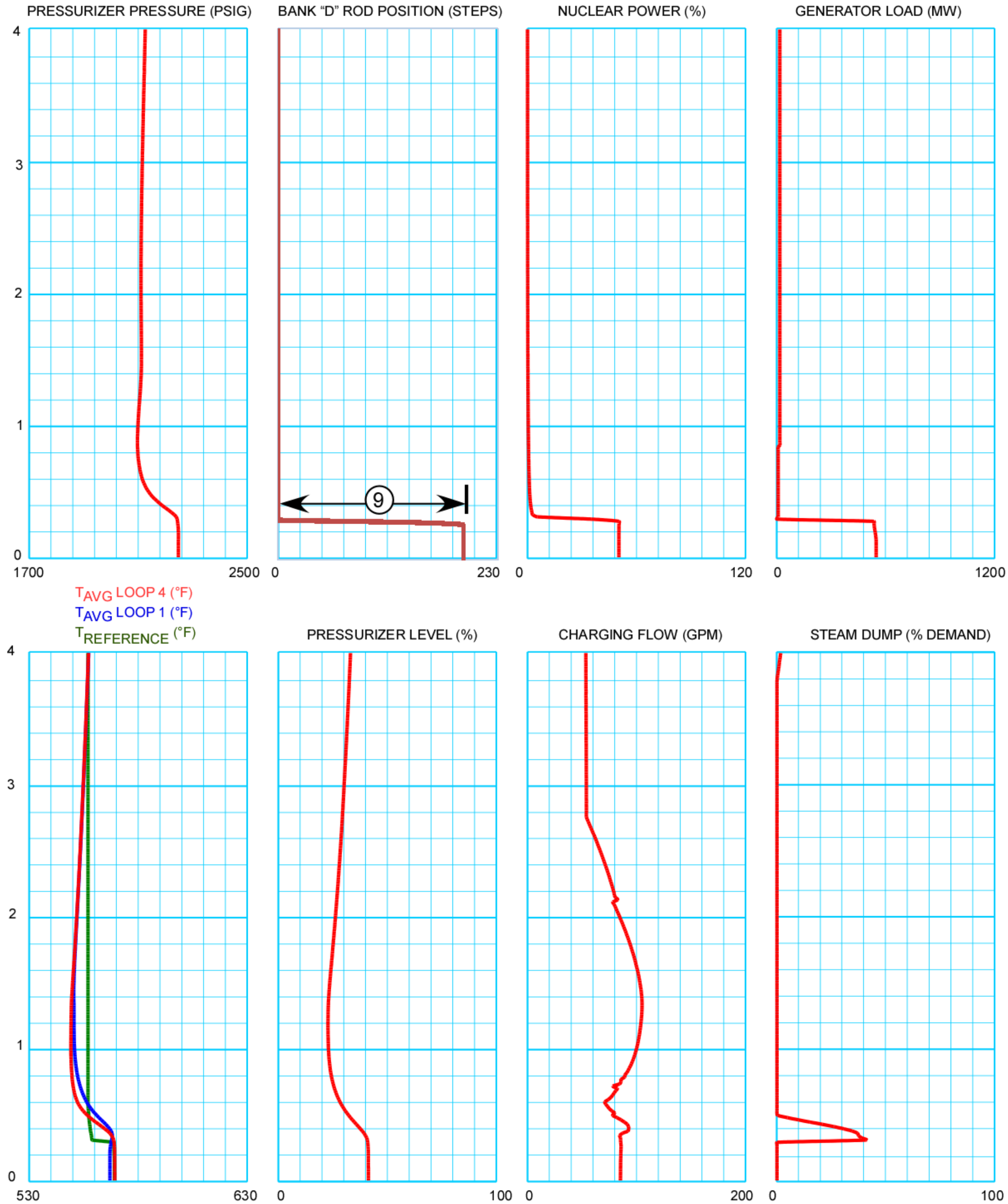
8. **Steam pressure in the #4 main steam line** (as well as in the #2 and #3 main steam lines) decreases in response to the increased heat transfer in (and increased steaming from) that SG. The 3 unaffected SGs are attempting to maintain the turbine load at the original level and cannot support the necessary steam flow at the original steam pressure.

9. The reactor trips (as indicated by the step drop in **bank D rod position**) on low-low steam generator water level in the #1 SG. The level shrink and reduction in feed flow quickly drop the water level off-scale low in that SG.

Transient 5.62 Inadvertent MSIV Closure (cont'd)

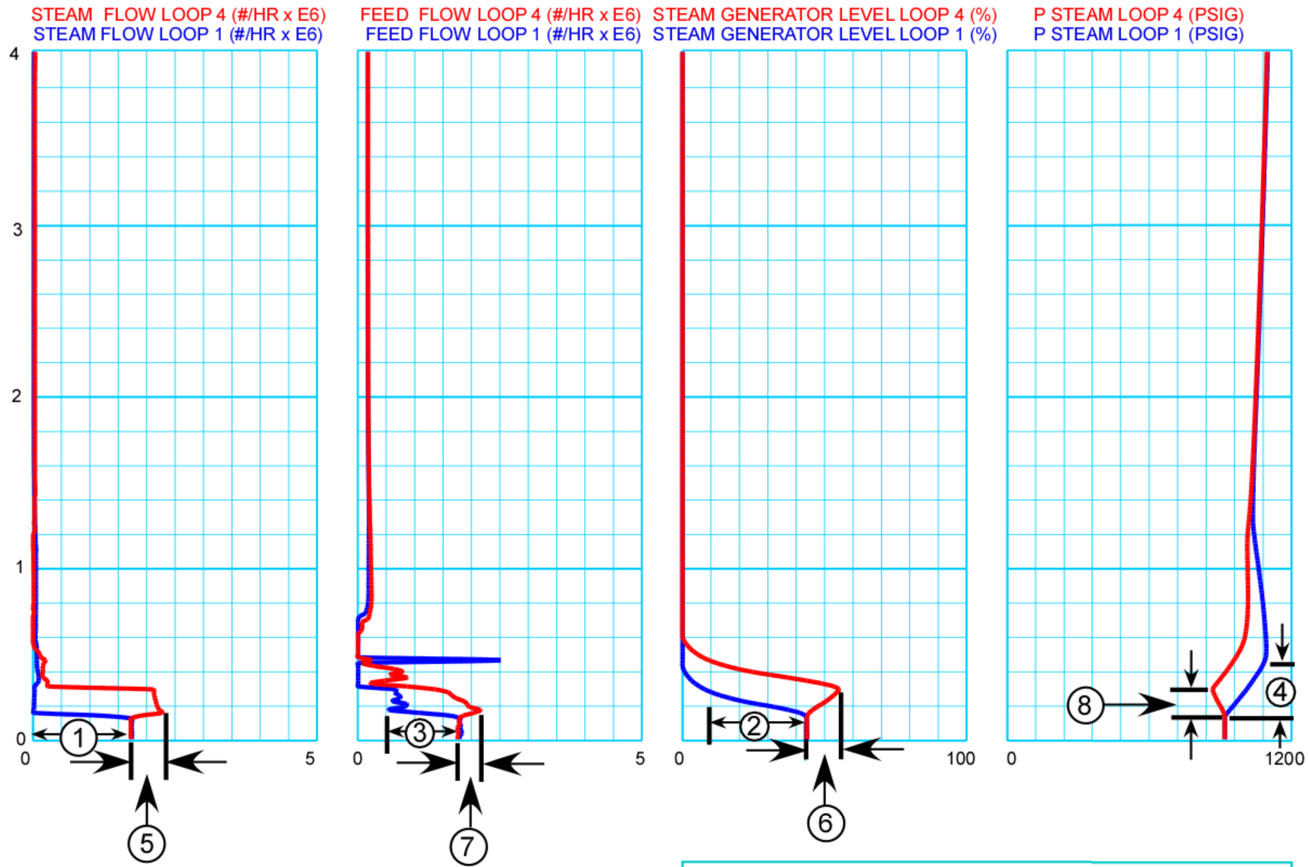
What this transient illustrates:

1. The effects of a closed MSIV: reduced steam flow, SG level shrink, increased steam pressure.
2. Increased steam flow from the other 3 main steam lines to "pick up the slack."



Transient 5.62 Inadvertent MSIV Closure

This page intentionally blank



TRANSIENT 5.62
INADVERTENT MSIV CLOSURE

Initial Conditions
BOL
Tavg: 567°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 50%
Bank D Rod Position: 194 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
The MSIV in the main steam line from the #1 SG inadvertently shuts

Transient 5.62 Inadvertent MSIV Closure

This page intentionally blank

Transient 5.63 RCP Trip

Initial Conditions:

BOL	Bank D Rod Position: 178 steps
T_{avg}: 563.2°F	Nuclear Power: 28%
PZR Pressure: 2235 psig	Feed reg. valve for SG #1 is in manual and maintaining ~60% level initially; it is placed in auto when the RCP is tripped.
	All other control systems in automatic

Initiating Event: The RCP in loop #1 trips

<u>Point</u> <u>Explanation</u>

1. **RCS flow in loop #1** decreases to 0 as the reactor coolant pump coasts down with flywheel inertia.
2. **RCS flow in loop #4** (as well as in loops #2 and #3) increases with the decrease in flow resistance from loop #1 and the development of reverse flow in that loop supplied by the discharge of the 3 running pumps.
3. The reduction in **steam flow in steam line #1** reflects the degradation of heat transfer in SG #1. The reduction of the reactor coolant flow rate in the tubes of that SG reduces the heat transfer coefficient for primary-to-secondary heat transfer, thereby reducing the boiling rate, and steam flow drops rapidly.
4. The relative absence of boiling in SG #1 and its associated flow resistance allows feedwater to flow into the tube bundle region from the downcomer (where SG level is measured), and the drop in **steam generator level in the #1 SG** results. Contributing to the level drop is the reduction in recirculation flow from the moisture separators to the downcomer with the reduction in steam flow. Note: the manual manipulation of SG level prior to the initiation of the transient prevents a reactor trip on low-low SG level.
5. The increase in **RCS flow in loop #1** reflects the development of reverse flow in that loop. Discharge from the 3 running pumps is supplied to the cold leg of the idle loop via the reactor vessel annulus.

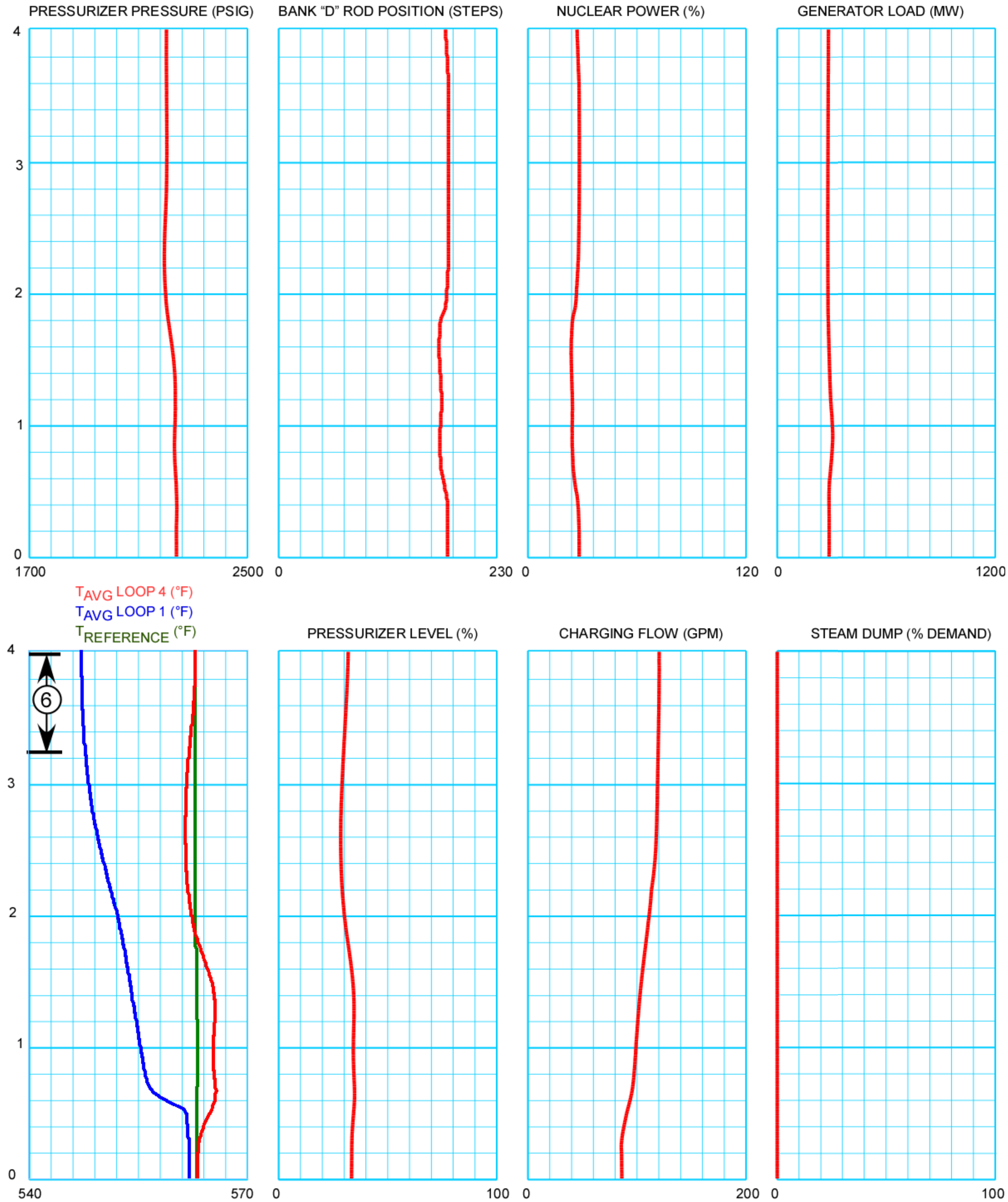
Point Explanation

6. The decreased value of **loop #1 T_{avg}** that exists at the end of the 4 plotted minutes also reflects the development of reverse flow in that loop. Once reverse flow in the idle loop is fully developed, the flow in that loop enters the SG (from the cold leg) with a temperature ~ equal to the T_c of the other 3 loops and exits (to the hot leg) a little colder. At this point, the greatly reduced flow and altered pressure drops in loop #1 mean that the actual loop conditions are transmitted to the bypass manifold RTDs with a much greater time lag and that the calculated T_{avg} may not be a true average loop temperature. However, the time span denoted by this point is a fairly long time after the development of reverse flow, and the actual loop #1 T_H and T_c differ by only a few degrees because of the minimal heat transfer in that loop's SG, so the indicated T_{avg} at the end of the 4 minutes should be very close to the transient endpoint T_{avg} in this loop.

7. **Steam flow in steam line #4** (as well as in steam lines #2 and #3) increases as heat transfer in the SGs in the loops with the running RCPs increases to maintain the unchanged turbine load. The increased steam flow is supported by a core ΔT which is a few degrees larger than its initial value now that the core mass flow rate has decreased to ~ 3/4 of its initial value ($\dot{Q} = \dot{M} c_p [T_H - T_c]$).

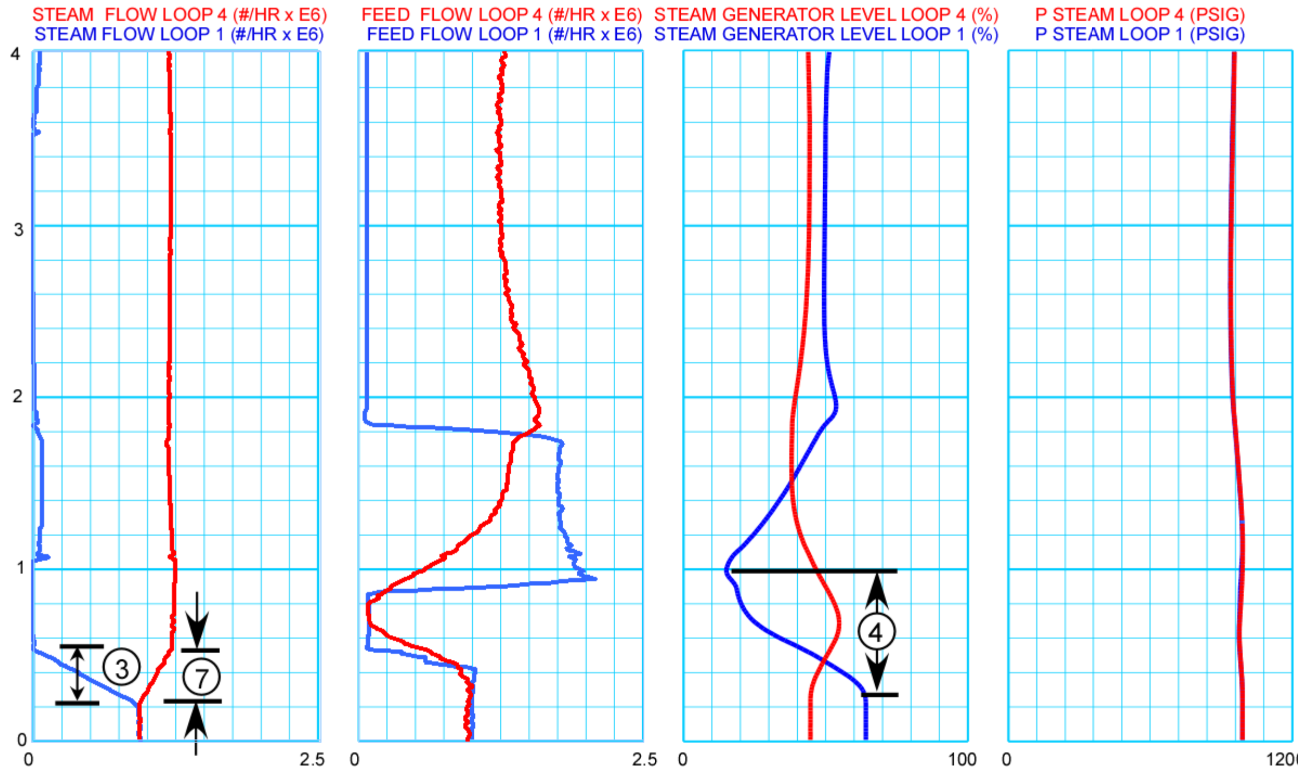
What this transient illustrates:

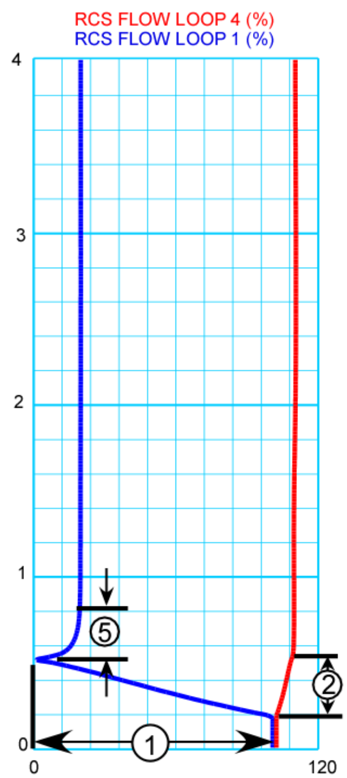
1. The reduction in RCS loop flow associated with an RCP trip and the subsequent development of reverse flow in that loop.
2. The shrink in SG level in the loop with the tripped RCP.



Transient 5.63 RCP Trip

This page intentionally blank





TRANSIENT 5.63
RCP TRIP

Initial Conditions
BOL
Tavg: 563.3°F
Pressurizer Pressure: 2235 psig
Nuclear Power: 28%
Bank D Rod Position: 178 steps
Charging flow provided by one centrifugal charging pump
The feed reg. valve for the #1 SG is in manual and maintaining 60% level initially; it is placed in auto when the RCP is tripped.
All other control systems are in automatic.

Initiating Event:
The RCP in loop #1 trips

Additional Information:
RCS Flow (Loops #1 and #4)

Transient 5.63 RCP Trip

This page intentionally blank

Transient 5.71 SG Safety Valve Fails Open

Initial Conditions:

BOL	Bank D Rod Position: 103 steps
T_{avg}: 558.2°F	Nuclear Power: ~10^{-8} amps in I.R.
PZR Pressure: 2235 psig	Normal plant configuration for startup

Initiating Event: One safety valve on main steam line #1 fails to 100% open

Point **Explanation**

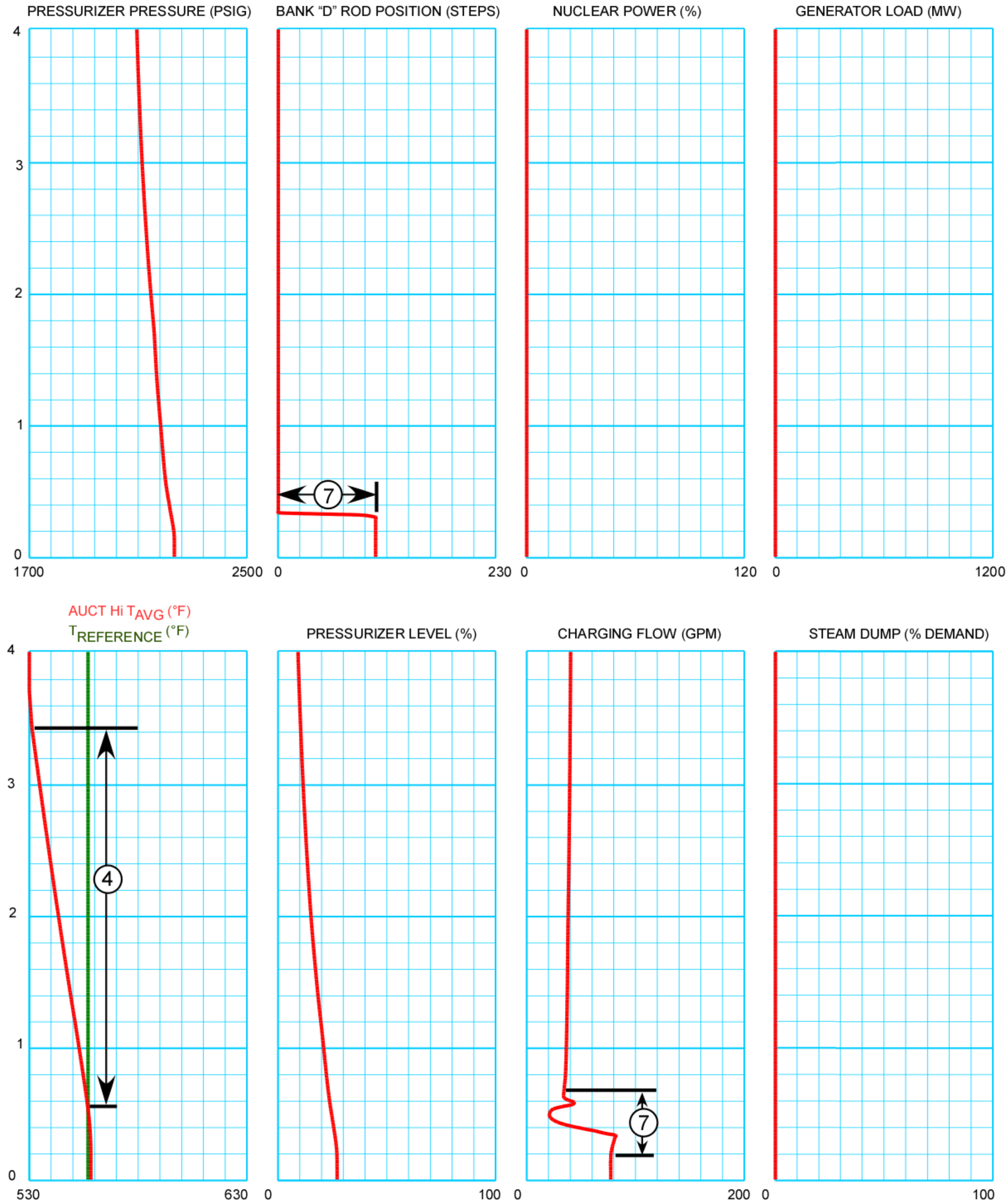
1. **Steam flow in main steam line #1** increases to the safety valve capacity for no-load steam pressure.
2. **Steam generator level in SG #1** swells with the surge in steam demand associated with the safety valve failing open.
3. **Steam pressure in main steam line #1** decreases as the failed safety valve discharges to the atmosphere.
4. **T_{avg}** decreases due to the cooldown of the reactor coolant caused by the additional steaming through the safety valve.
5. **Steam pressure in main steam line #4** (as well as the other intact main steam lines) decreases but lags the steam pressure in main steam line #1. The other three main steam lines cannot feed the failed-open safety valve because the check valve in main steam line #1 prevents backflow, but the cooldown of the reactor coolant (see explanation for point 4 above) is reducing the steam pressures in the intact steam lines.
6. **Intermediate range power** increases in response to the positive reactivity insertion associated with the decrease in reactor coolant temperature. The increase is rather small: 2.47×10^{-8} amps to 2.71×10^{-8} amps.
7. The reactor trip (indicated by the step drop in **bank D rod position**) and ESF actuation (indicated by the characteristic perturbation in **charging flow**) is caused by high steam line ΔP. At ~19 sec, the ΔP between steam line #1 and at least two of the other steam lines has reached the 100 psi setpoint.
8. The decrease in **steam generator level in SG #1** reflects dissipation of that SG's inventory through the failed-open safety valve.

Point **Explanation**

9. **Feed flow** during the latter few minutes of the transient reflects AFW system operation. The AFW system actuation is caused by the ESF actuation. The AFW flow to SG #1 is a little larger than that to the other 3 SGs because of the lower pressure in SG #1.

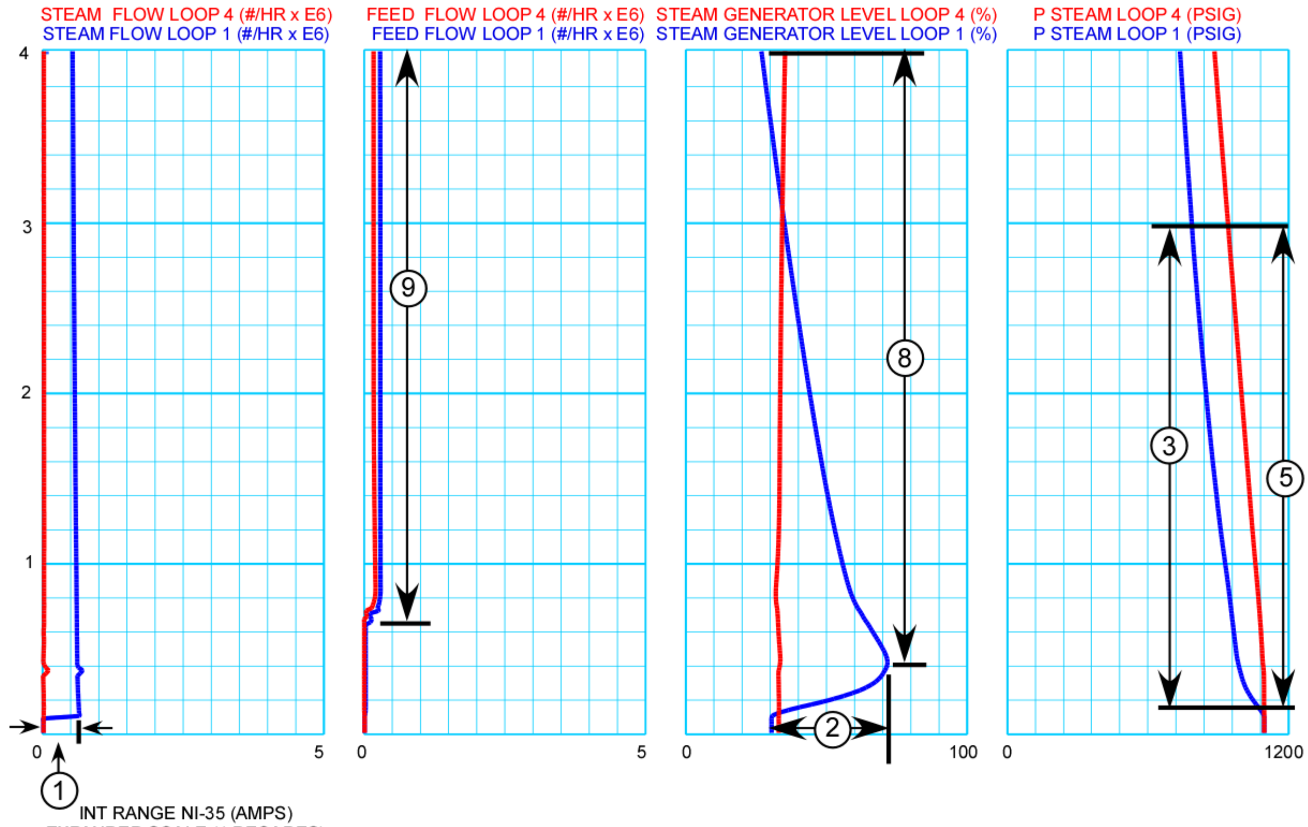
What this transient illustrates:

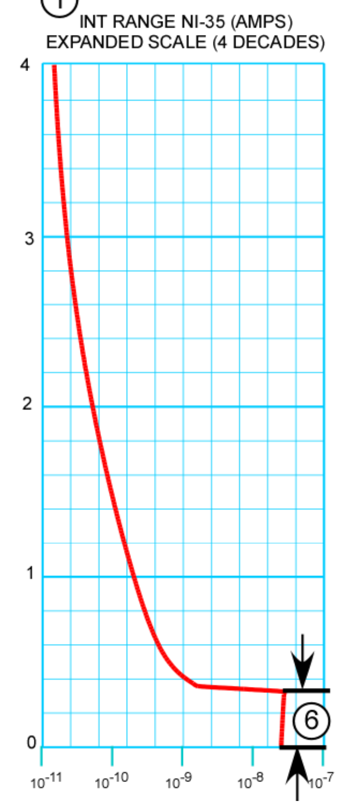
1. The relatively slow dissipation of one SG's inventory through a failed-open safety valve.
2. The isolation of the steam break from the other 3 SGs by the faulted SG's check valve.
3. A steam-break-induced cooldown of the RCS.
4. An increase in reactivity associated with a decrease in RCS temperature.
5. A high steam line ΔP ESF actuation and the characteristic response of charging flow.



Transient 5.71 SG Safety Valve Fails Open

This page intentionally blank





TRANSIENT 5.71
SG SAFETY VALVE FAILS OPEN

Initial Conditions
BOL
Tavg: 558.5°F
Pressurizer Pressure: 2233 psig
Nuclear Power: ~10^{-8} amps in I.R.
Bank D Rod Position: 103 steps
Charging flow provided by one centrifugal charging pump
Normal plant configuration for startup.

Initiating Event:
One safety valve on main steam line #1 fails to 100% open

Additional Information:
Intermediate Range Power (expanded scale)

Transient 5.71 SG Safety Valve Fails Open

This page intentionally blank

Transient 5.72 Large Steam Break Inside Containment With Loop, 10^{-8} Amps In I.R.

Initial Conditions:

BOL	Bank D Rod Position: 103 steps
T_{avg}: 558.2°F	Nuclear Power: ~10^{-8} amps in I.R.
PZR Pressure: 2232 psig	Normal plant configuration for startup

Initiating Event: Large break on main steam line #1 (8.72 X 10^6 lbm/hr) and LOOP occur simultaneously at ~5 sec

Point **Explanation**

1. **Steam flow in main steam line #1** increases off-scale high with the steam break.
2. **Steam generator level in SG #1** swells off-scale high with the surge in steam demand associated with the steam break. Note that the wide range channel essentially does not respond to shrink and swell.
3. **Steam pressure in main steam line #1** decreases as steam is discharged to the containment atmosphere.
4. **Steam pressure in main steam line #4** (as well as the other intact main steam lines) remains at or near its initial value. The other three main steam lines cannot feed the break because the check valve in main steam line #1 prevents backflow.
5. **Wide-range loop #1 T_c** decreases due to the cooldown of the reactor coolant caused by the steam break, which is in steam line #1 and affects loop #1 the most.
6. **Charging flow** decreases to 0 very rapidly with the LOOP; the CCP which has been operating is now unpowered.
7. The reactor trip (indicated by the step drop in **bank D rod position**) is caused by the SG ∆P safety injection actuation. The loss of offsite power would also cause the rods to drop, but the large flywheels on the rod drive motor generators keep the stationary grippers engaged for about 9 seconds after a loss of power.
8. **Charging flow** returns after the emergency diesel generators have started and attained rated speed, the EDG output breakers have closed, and the CCPs have restarted. The CCPs are the first pieces of emergency equipment to be started by the DBA sequencer, which is initiated by a high steam line ∆P safety injection actuation. (The ∆P between steam line #1 and at least two of the other steam lines reaches the 100 psi setpoint about 3 sec after the start of the accident.) The CCPs are now operating in the HPI mode; the plotted charging flow reflects seal injection to the RCPs (normal charging is isolated) and a constant charging flow control valve position (see the note below).

Transient 5.72 Large Steam Break Inside Containment With Loop, 10^{-8} Amps In I.R. (cont'd)

Point Explanation

9. **Feed flow to SG #1** increases with actuation of the AFW system (the AFW system is actuated by the ESF actuation). The flow to that SG is initially higher than that to the other three SGs because of its lower pressure. The AFW flow to SG #1 then decreases when each AFW flow control valve to that SG closes in response to the high flow (> 500 gpm) sensed in its supply line. The train B flow control valve closes before the train A valve does, hence the "step-down" nature of the AFW flow reduction.

10. The decrease in **steam flow in main steam line #1** reflects the decrease in driving force for steam flow with the reduction in steam pressure in that steam line and the increase in containment pressure.

11. The decrease in **steam generator level in SG #1** reflects the rapid dissipation of that SG's inventory through the steam break.

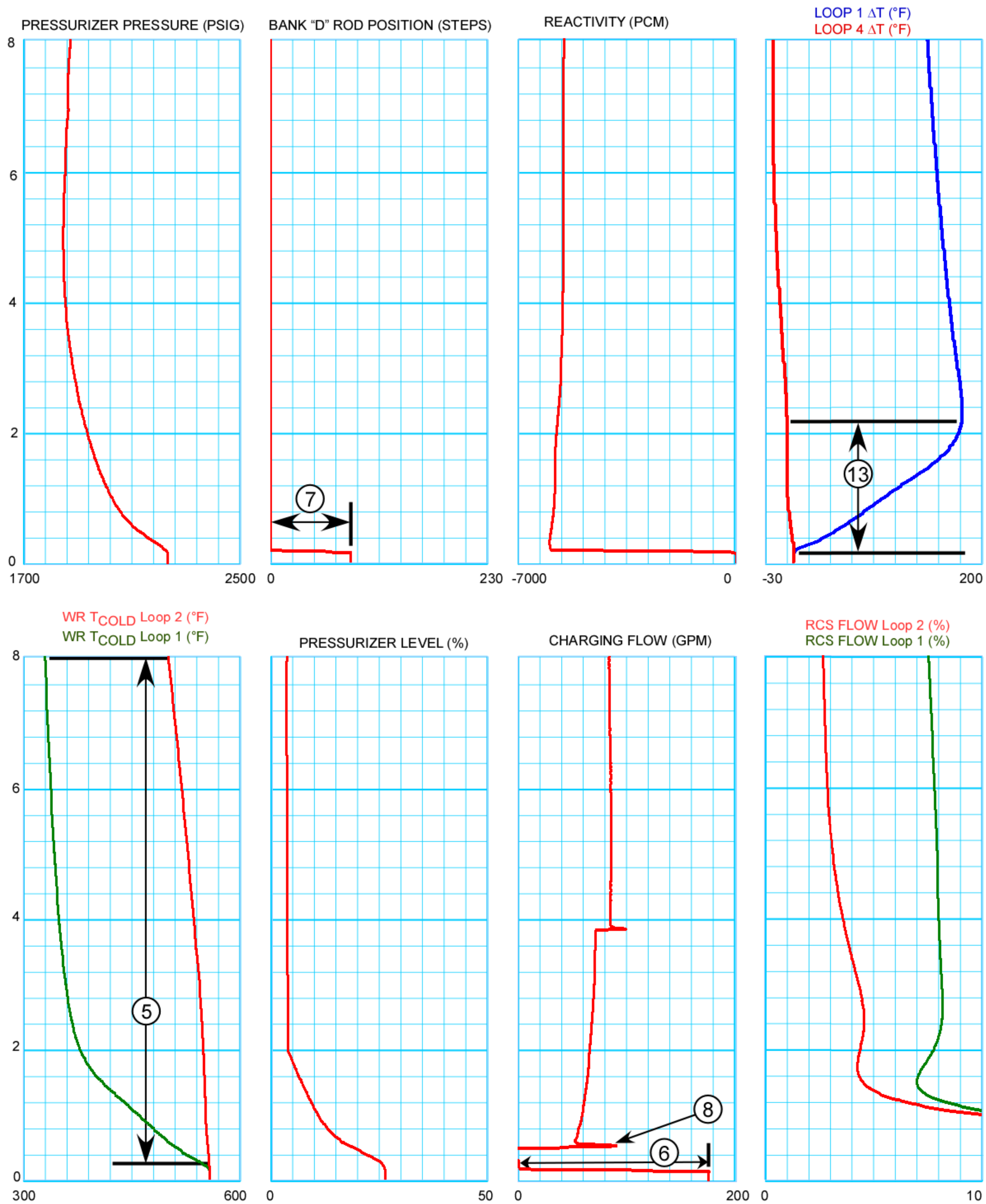
12. **Containment pressure** increases as steam is discharged from main steam line #1 to the containment atmosphere. The pressure increase is tempered by containment fan cooler operation.

13. Loop #1 ΔT rises to a large value due to the high steaming rate. Loop #4 ΔT reverses as the intact steam generators become a heat source for the RCS. The #1 SG is cooling the RCS, which in turn cools down the intact steam generators.

Note : Upon momentary loss of power, the controller for the charging flow control valve (FCV-121) shifts to manual and holds the valve at the last automatically demanded position (partly closed). At about the 4 minute point, FCV-121 fails open as instrument air pressure is lost due to the loss of air compressors.

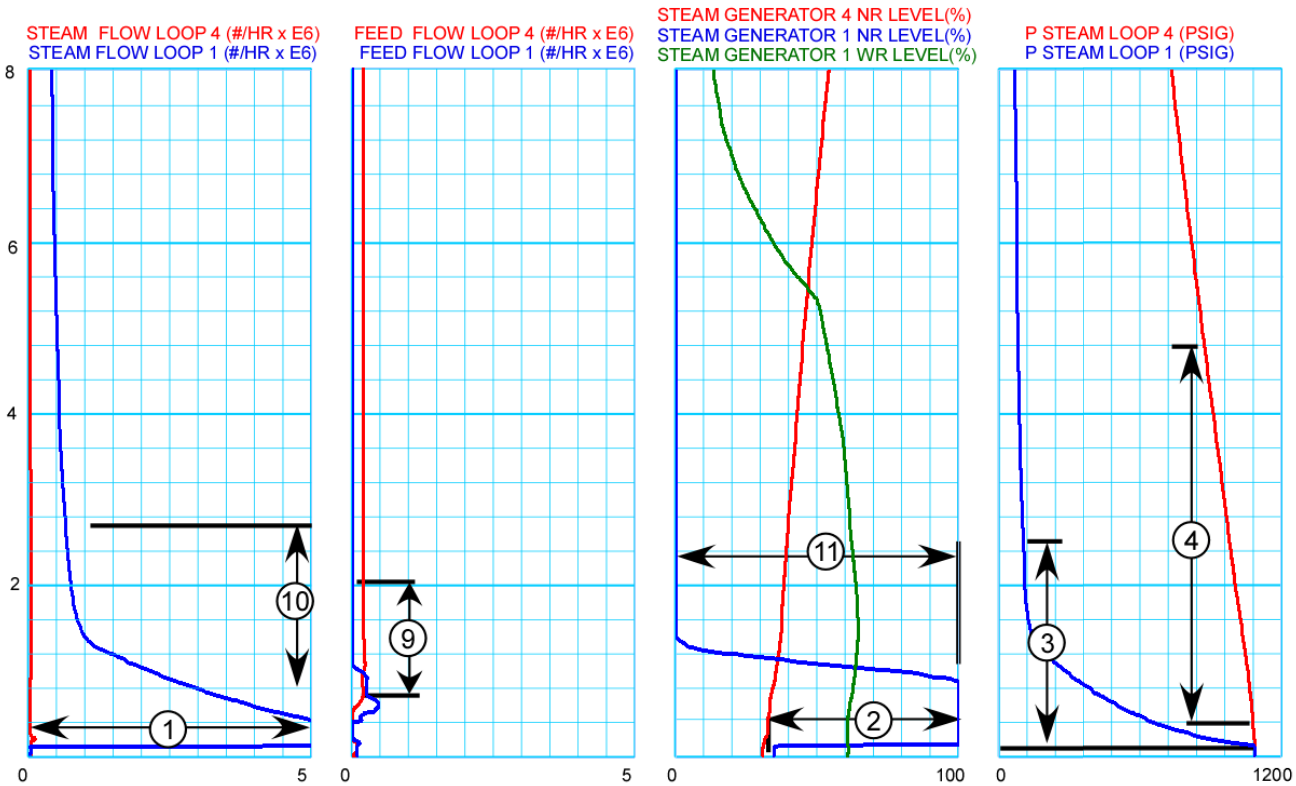
What this transient illustrates:

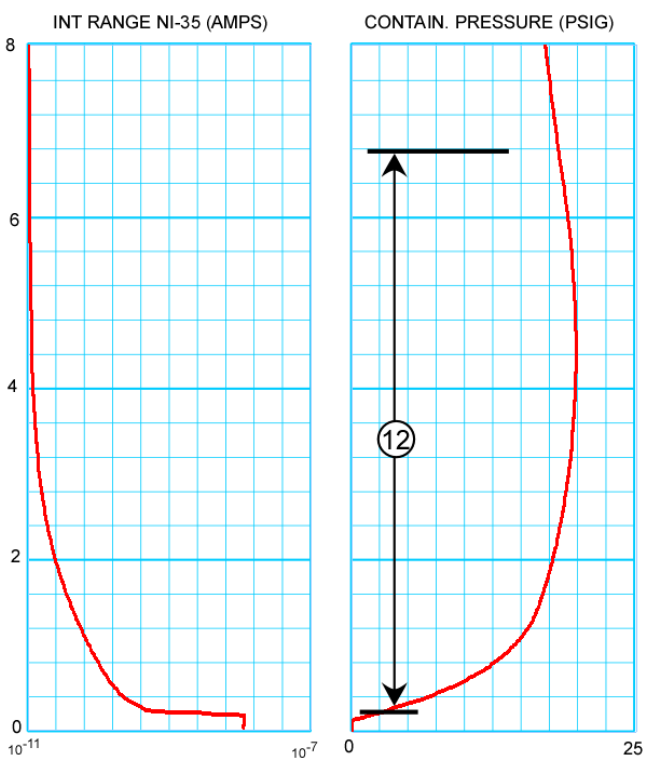
1. The dissipation of one SG's inventory through an unisolable break upstream of the MSIV.
2. The isolation of the steam break from the other 3 SGs by the faulted SG's check valve.
3. A steam-break-induced cooldown of the RCS.
4. A high steam line ΔP ESF actuation and the characteristic response of charging flow.
5. LOOP-induced effects: the delay in starting the charging pumps after the ESF actuation and the controller responses.
6. The "feed-only-good-generator" feature of the AFW flow control valves.
7. The intact steam generators become heat sources to the RCS during the steam-break-induced cooldown.



Transient 5.72 Large Steam Break Inside Containment With LOOP, 10^{-8} Amps In I.R.

This page intentionally blank





TRANSIENT 5.72
LARGE STEAM BREAK INSIDE CONTAINMENT
WITH LOSS OF OFFSITE POWER

Initial Conditions
BOL
Tavg: 558.5°F
Pressurizer Pressure: 2247 psig
Nuclear Power: ~10^{-8} amps in I.R.
Bank D Rod Position: 103 steps
Charging flow provided by one centrifugal charging pump
Normal plant configuration for startup.

Initiating Event:
Large break on main steam line #1 (8.72×10^6 lbm/hr) and loss of offsite power occur sumultaneously.

Transient 5.72 Large Steam Break Inside Containment With LOOP, 10^{-8} Amps In I.R.

This page intentionally blank

Transient 5.73 Large Steam Break Inside Containment, 100% Power

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.2°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	All control systems in automatic

Initiating Event: Large break on main steam line #1 (8.72×10^6 lbm/hr) inside containment

Point **Explanation**

1. **Steam flow in main steam line #1** increases off-scale high with the steam break.

2. **Steam generator level in SG #1** swells with the surge in steam demand associated with the steam break. Note that the wide range level does not swell with increasing steam flow.

3. **Steam pressure in main steam line #1** decreases as steam is discharged to the containment atmosphere.

4. The removal of flow resistance associated with the isolation of the #1 main steam line (because of closure of its check valve) causes **steam flow from the #4 SG** (as well as from the #2 and #3 SGs) to increase.

5. The reactor trip (indicated by the step drop in **bank D rod position**) and ESF actuation (indicated by the characteristic perturbation in **charging flow**) is caused by high steam flow plus low steam pressure. The high steam flow setpoint is exceeded with the break in steam line #1 and the increased flow in the other 3 steam lines (see the explanation for point 4 above), and the low steam pressure setpoint is reached as the steam pressure decreases in all steam lines. Note that the ESF actuation takes place when the steam pressure is ~ 750 psig in the intact steam lines; the input to the low steam pressure bistable reaches the low steam pressure setpoint before actual pressure does because of the lead-lag circuit through which the steam pressure signal is processed. Note also that the other ESF actuation signals are not possible: pressurizer pressure and T_{avg} are not low enough yet, and steam line ΔP and containment pressure are not large enough yet. However, one other reactor trip signal is possible: a turbine trip + P-7, with the turbine tripping on high SG level.

6. The closure of the steam line check valves and MSIVs allows **steam pressure in main steam line #4** (as well as in steam lines #2 and #3) to recover following the reactor trip and ESF actuation.

7. **Wide-range loop #1 T_c** decreases due to the cooldown of the reactor coolant caused by the steam release through the break. The cooldown in RCS loop #1 (which contains the SG which is feeding the break) leads the cooldowns in the other loops. The cooldown stops when the steam generator is more or less empty.

<u>**Point**</u> <u>**Explanation**</u>

8. **Feed flow to SG #1** increases with actuation of the AFW system (the AFW system is actuated by the ESF actuation). The flow to that SG is initially higher than that to the other three SGs because of its lower pressure. The AFW flow to SG #1 decreases to 0 in two steps when each AFW flow control valve to that SG closes in response to the high flow (> 500 gpm) sensed in its supply line to that SG. The two steps are due to the train A and train B valves closing at different times.

9. The decrease in **steam flow in main steam line #1** reflects the decrease in driving force for steam flow with the reduction in steam pressure in that steam line and the increase in containment pressure.

10. The decrease in **steam generator level in SG #1** reflects the rapid dissipation of that SG's inventory through the steam break.

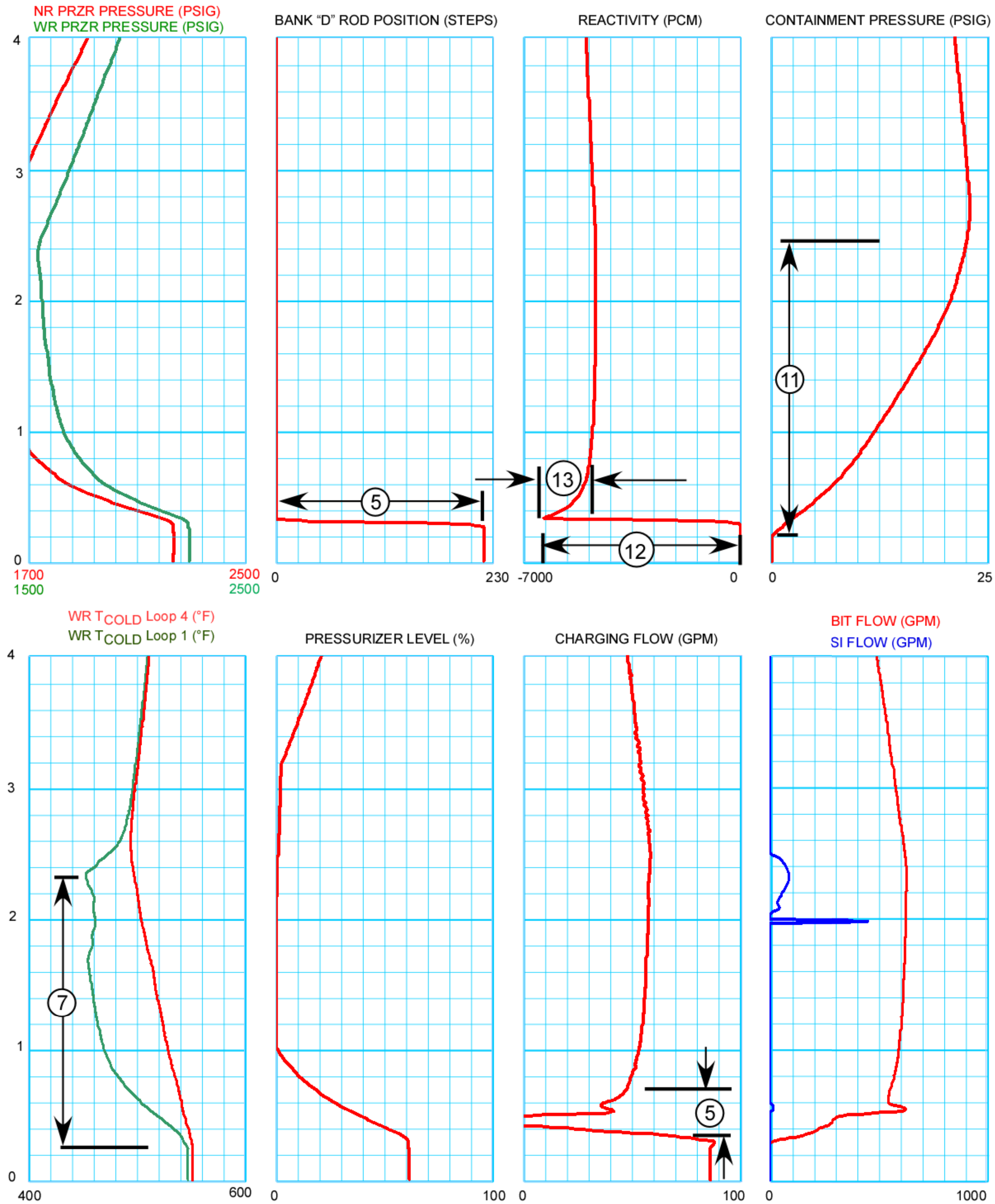
11. **Containment pressure** increases as steam is discharged from main steam line #1 to the containment atmosphere. The pressure increase is tempered by containment fan cooler operation.

12. The reactor is made subcritical by the amount of total rod worth minus power defect.

13. The positive reactivity added by the RCS cooldown is mitigated by ECCS flow.

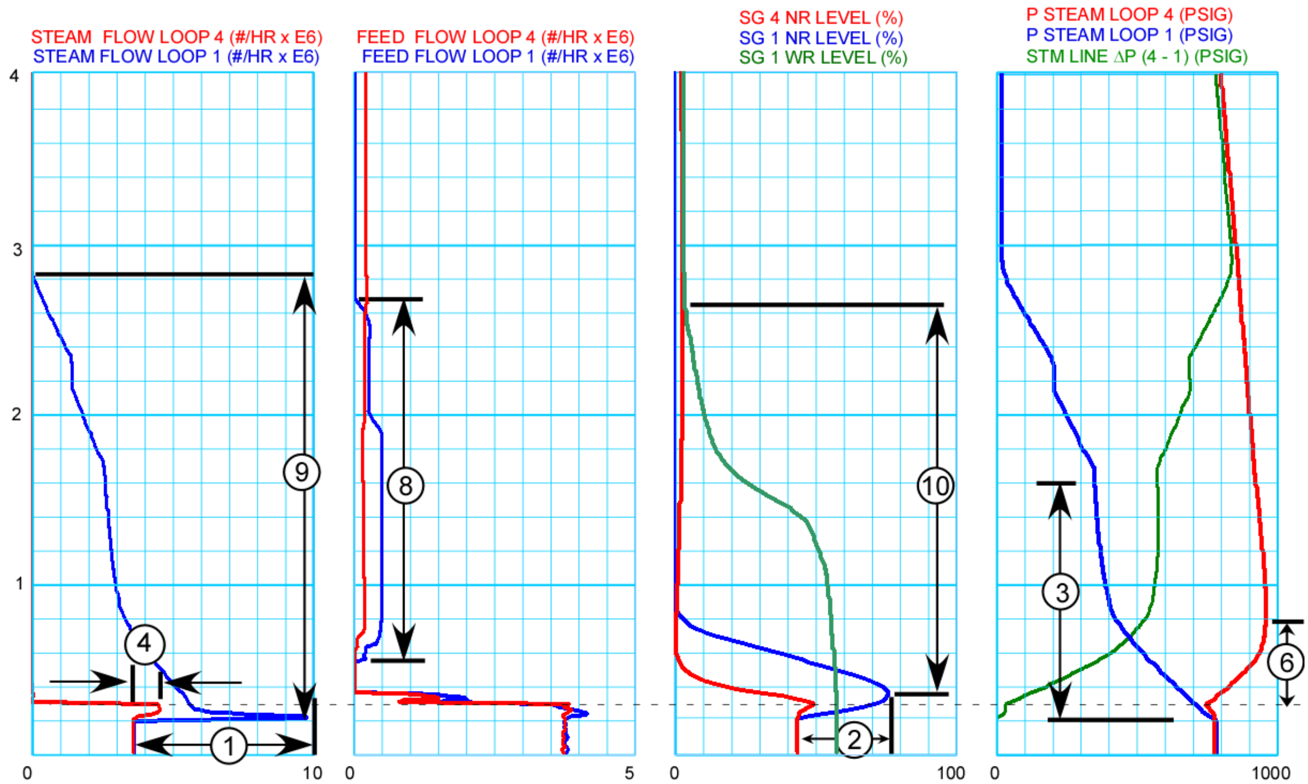
What this transient illustrates:

1. The dissipation of one SG's inventory through an unisolable break upstream of the MSIV.
2. The isolation of the steam break from the other 3 SGs by the faulted SG's check valve and by MSIV closure.
3. A steam-break-induced cooldown of the RCS.
4. A high steam flow + low steam pressure ESF actuation and the characteristic response of charging flow.
5. The "feed-only-good-generator" feature of the AFW flow control valves.



Transient 5.73 Large Steam Break Inside Containment, 100% Power

This page intentionally blank



TRANSIENT 5.73
LARGE STEAM BREAK INSIDE CONTAINMENT

Initial Conditions
BOL
Tavg: 585.2°F
Pressurizer Pressure: 2245 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
Large break on main steam line #1 (8.72×10^6 lbm/hr)

Additional Information:
Reactivity
Containment pressure
ECCS flow

Transient 5.73 Large Steam Break Inside Containment, 100% Power

This page intentionally blank

Transient 5.74 Large Steam Break Downstream Of MSIVs, 10^{-8} Amps In I.R.

Initial Conditions:

BOL	Bank D Rod Position: 103 steps
T_{avg}: 558.2°F	Nuclear Power: ~10^{-8} amps in I.R.
PZR Pressure: 2232 psig	All control systems in automatic

Initiating Event: Large break (8.72 X 10^6 lbm/hr) downstream of the MSIVs

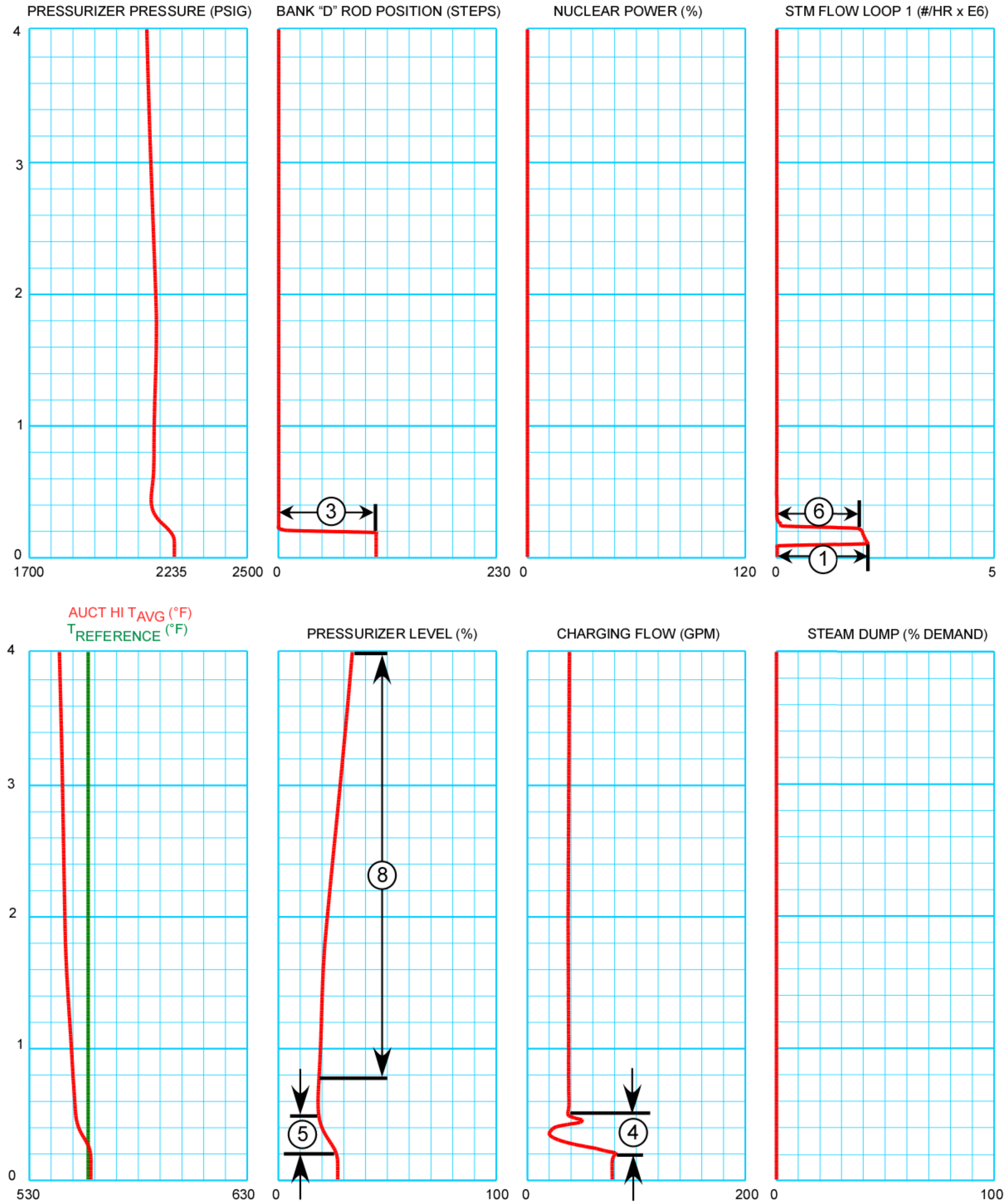
<u>**Point**</u> <u>**Explanation**</u>

1. **Steam flow** (all main steam lines) increases rapidly with the steam break.
2. **Steam pressure** decreases with the unrestricted steam flow through the break.
3. The reactor trip (indicated by the step drop in **bank D rod position**) is caused by an ESF actuation on high steam flow plus low steam pressure. The high steam flow setpoint is reached with the increase in steam flow through the break (the high steam flow setpoint is at its minimum value with no turbine load), and the low steam pressure setpoint is reached as the steam pressure constantly decreases. Note that the ESF actuation takes place when the steam pressure is ~ 980 psig; the input to the low steam pressure bistable reaches the low steam pressure setpoint before actual pressure does because of the lead-lag circuit through which the steam pressure signal is processed. Note also that the other ESF actuation signals are not possible: pressurizer pressure and T_{avg} are not low enough yet, and, because the break is downstream of the MSIVs, there is no steam line ΔP, and it does not cause containment pressure to increase.
4. The **charging flow** perturbation reflects the ESF actuation (isolation of normal charging and diversion of some HPI flow through the seal injection lines).
5. The additional steam flow through the break cools the reactor coolant; the **pressurizer level** decrease reflects the reduction in reactor coolant volume caused by the decrease in coolant temperature.
6. **Steam flow** decreases to 0, reflecting the MSIV closure associated with the high steam flow ESF actuation.
7. **Source range power** comes on scale when both source range detectors are energized by both I.R. channels reaching the P-6 setpoint (10^{-10} amps) following the trip.
8. **Pressurizer level** increases with injection from the CCPs and letdown isolated following the ESF actuation.

Transient 5.74 Large Steam Break Downstream Of MSIVs, 10^{-8} Amps In I.R. (cont'd)

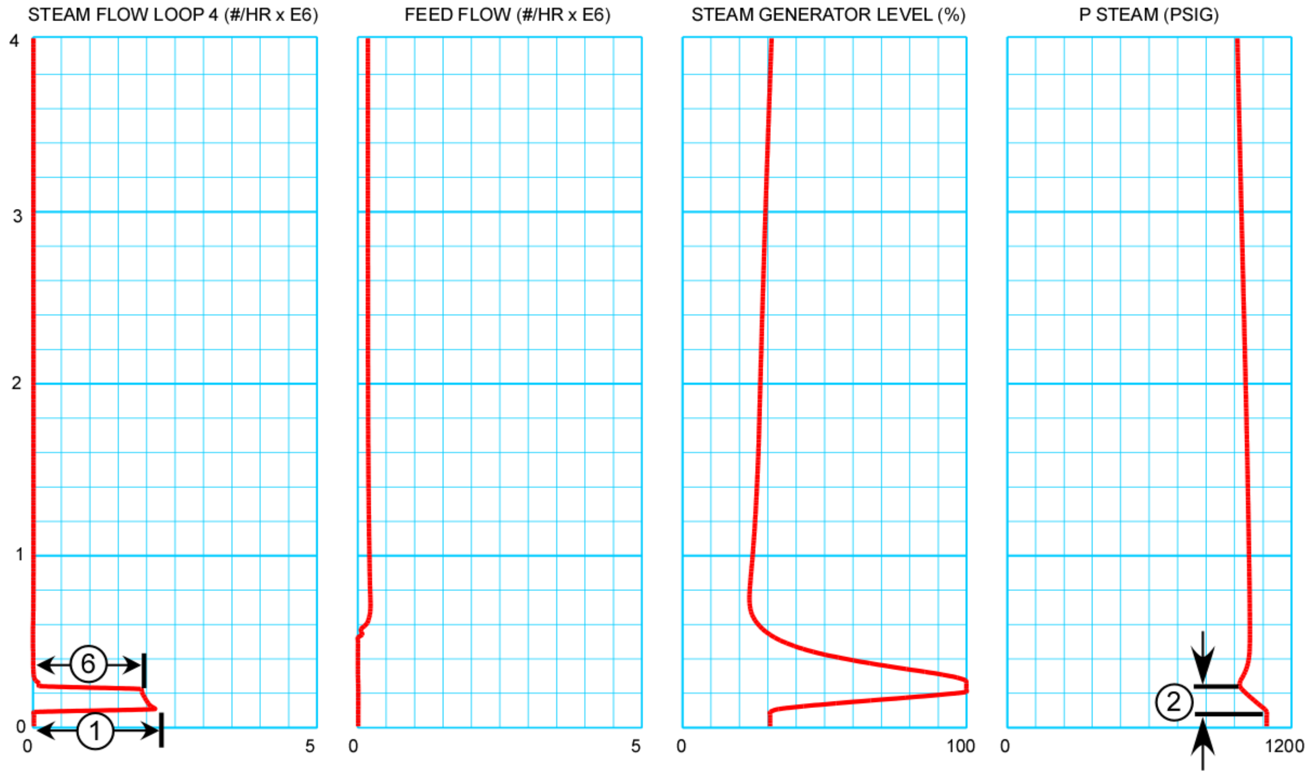
What this transient illustrates:

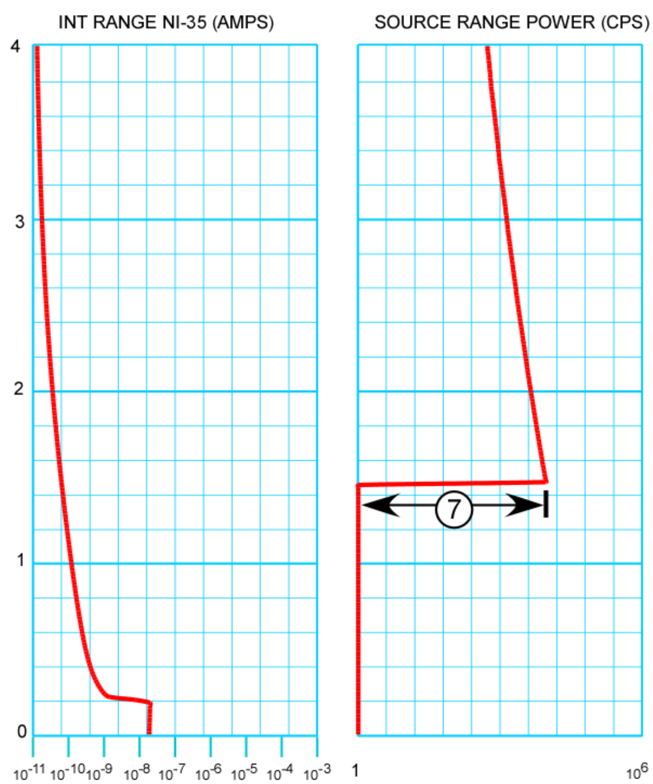
1. The feeding of a steam break downstream of the MSIVs by all SGs.
2. A high steam flow + low steam pressure ESF actuation and the isolation of all main steam lines.
3. A steam-break-induced cooldown of the RCS.
4. The energizing of the source range detectors when intermediate range power reaches the P-6 setpoint.



Transient 5.74 Large Steam Break Downstream Of MSIVs, 10^{-8} Amps In I.R.

This page intentionally blank





TRANSIENT 5.74
LARGE STEAM BREAK DOWNSTREAM OF MSIVs

Initial Conditions
BOL
Tavg: 558.1°F
Pressurizer Pressure: 2247 psig
Nuclear Power: ~10^{-8} amps in I.R.
Bank D Rod Position: 103 steps
Charging flow provided by one centrifugal charging pump
Normal plant configuration for startup

Initiating Event:
Large break (8.72 x 106 lbm/hr) downstream of the MSIVs

Additional Information:
Intermediate range power
Source Range Power
Loop 1 steam flow (replaces generator load)

Transient 5.74 Large Steam Break Downstream Of MSIVs, 10^{-8} Amps In I.R.

This page intentionally blank

Transient 5.75 SGTR In SG #1

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.3°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	All control systems in automatic

Initiating Event: 550-gpm tube rupture in SG #1

Point **Explanation**

1. **Pressurizer level** drops with the loss of coolant inventory from the RCS.
2. **Pressurizer pressure** decreases with the expansion of the pressurizer steam bubble associated with the inventory loss.
3. **Charging flow** increases to a very high value as the pressurizer level control system opens charging flow control valve FCV-121 in response to the low level, and the discharge flow from the operating CCP is enhanced by the low RCS pressure.
4. **Steam generator level in SG #1** increases by a few percent with the leakage of reactor coolant into that SG. The level is brought back to setpoint as the level error input to the SG water level control system reduces feed flow to SG #1 (see point 5 below).
5. **Feed flow to SG #1** decreases in response to the level error input to the SG water level control system.
6. **Generator load** decreases in response to OTΔT runbacks. The OTΔT runback sepoint is greatly reduced by the decrease in pressurizer pressure. A close inspection of the plot reveals 3 separate runbacks. In the turbine EHC system each runback is accomplished by the reduction of the load demand by ~ 5% in a 2.3-sec interval; runbacks continue every 30 sec as long as the runback condition persists.
7. The reactor trips (indicated by the step drop in **bank D rod position**) on low pressurizer pressure. OTΔT is a possible cause of the trip, but did not initiate the trip in this transient.
8. The low pressurizer pressure (1807-psig setpoint) ESF actuation (as indicated by the perturbation in **charging flow**) occurs a few seconds after the reactor trip. The shift in the charging flow value is exaggerated because the beginning and ending charging flows are so high due to the large charging demand (FCV-121 is wide open) and low RCS pressure.

Point Explanation

9. **Pressurizer pressure** decreases even faster after the pressurizer empties. Also contributing to the pressure drop at this point are the continued inventory loss and coolant volume contraction as T_{avg} is brought to the no-load value. The minimum pressure reached is ~ 1090 psig.

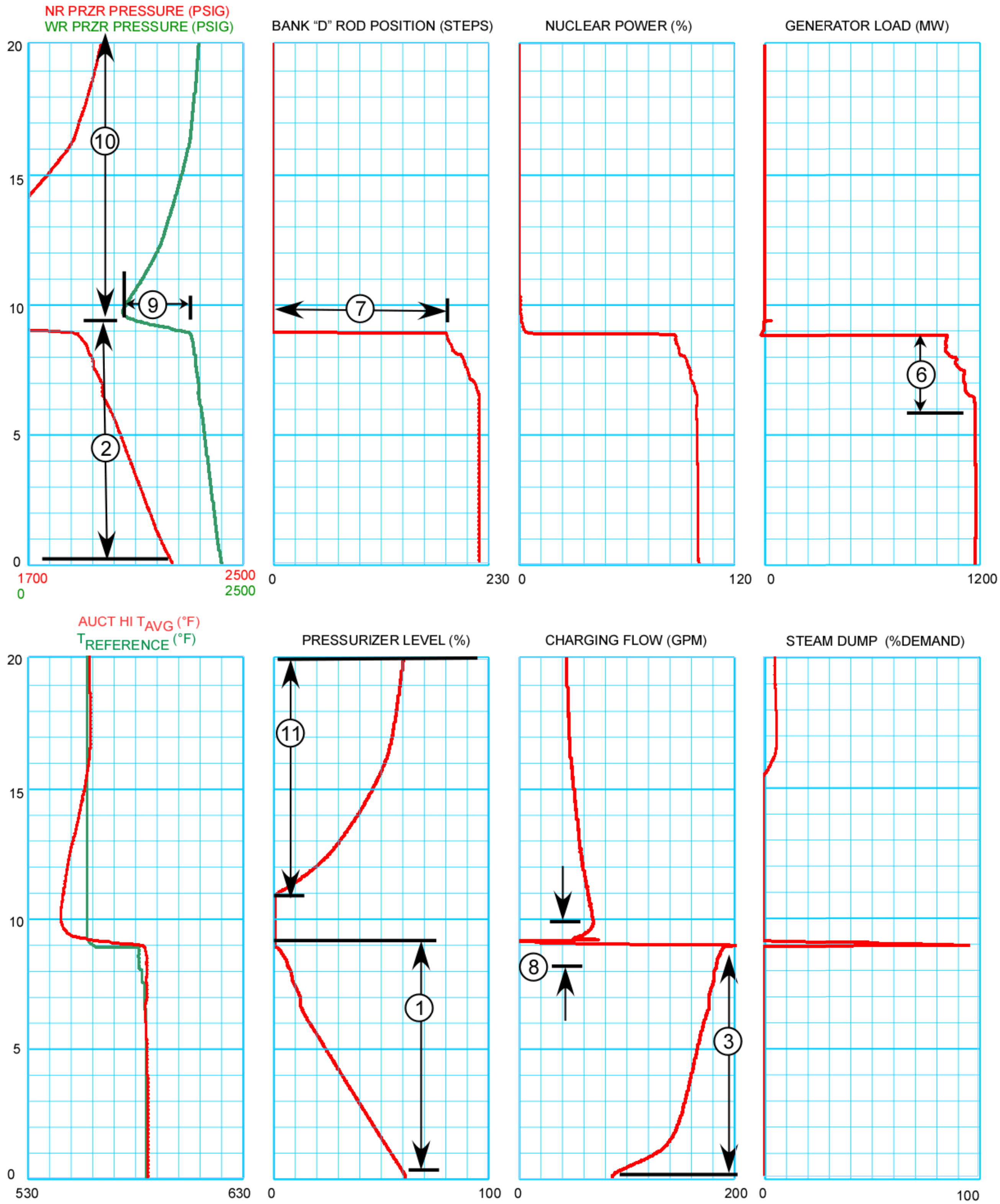
10. **Pressurizer pressure** recovers as total ECCS flow from the CCPs and the safety injection pumps exceeds the flow through the tube rupture. The decrease in RCS pressure has both increased injection flow and reduced the driving force for tube leakage. Existing steam bubbles in the RCS are being squeezed.

11. The increase in **pressurizer level** reflects the recovery of coolant inventory with ECCS flow > tube leakage. Since the rise in pressurizer level causes pressurizer pressure to rise, which causes RCS leak flow to rise and ECCS flow to drop, pressurizer level is approaching an equilibrium where RCS leak flow will equal ECCS flow.

12. **Steam generator level in SG #1** comes back on scale first because it is filling with AFW flow and reactor coolant leakage, while the other 3 SGs are filling with AFW flow alone.

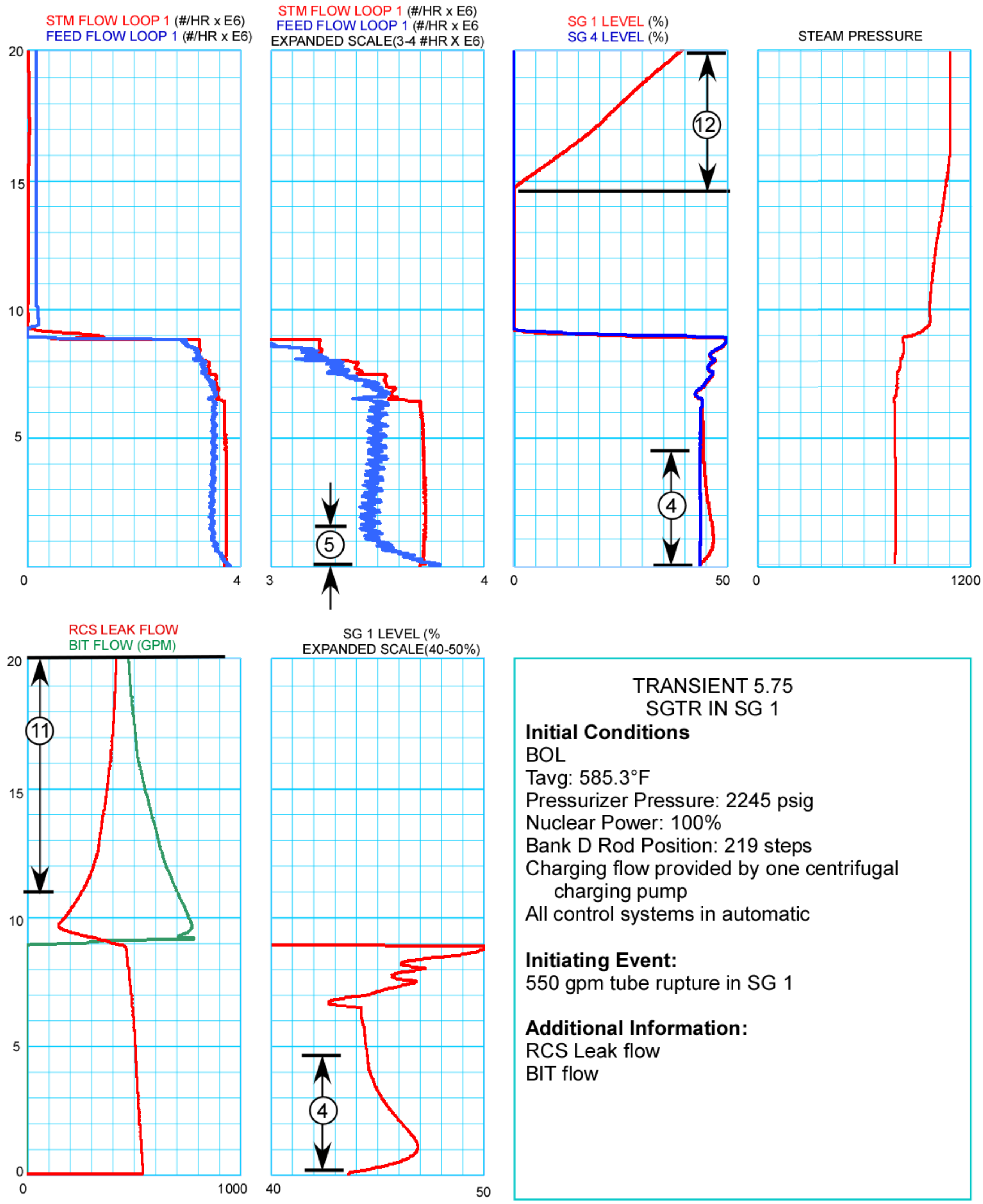
What this transient illustrates:

1. The loss of inventory from the RCS and the reduction in RCS pressure associated with an SGTR, and the resulting protection system responses.
2. A low pressurizer pressure ESF actuation and the characteristic response of charging flow.
3. OTΔT runbacks.
4. The filling of the RCS and the SGs by ESF systems.
5. Pressurizer level reaches an equilibrium after the trip where RCS leak flow = ECCS flow.



Transient 5.75 SGTR In SG #1

This page intentionally blank



Transient 5.75 SGTR In SG #1

This page intentionally blank

Transient 5.76 6-IN. Cold-Leg Break

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.2°F	Nuclear Power: 100%
PZR Pressure: 2232 psig	All control systems in automatic

Initiating Event: 40,000-gpm (~6-in. diameter) break in cold leg of loop #1

Failure of fast transfer to offsite power when the main generator trips

<u>Point</u> **<u>Explanation</u>**

1. **Pressurizer level** drops rapidly with the loss of coolant inventory from the RCS.

2. **Pressurizer pressure** decreases rapidly with the expansion of the pressurizer steam bubble and then complete emptying of the pressurizer.

3. The reactor trips (indicated by the negative reactivity and bank "D" rod position) on low pressurizer pressure or OTΔT. Both setpoints are reached during a rapid RCS depressurization.

4. **Core flow** drops to a very small value due to the loss of offsite power. Natural circulation does not develop because the steam generators are not the heat sink (see point 6).

5. **Pressurizer pressure** "hangs up" at ~ 940 psig. The saturation temperature for this pressure is ~ 540°F; a great deal, if not all, of the reactor coolant has reached saturation. This means that flashing is occurring at many locations in the RCS, and the formation of steam bubbles holds up the pressure decrease.

6. **RCS pressure** drops below the pressure of the steam generators, indicating that the steam generators are hotter than the RCS. The steam generators are a heat source, not a heat sink. Decay heat is being removed by a combination of break flow and ECCS flow.

7. **Break flow** drops as the coolant inventory at the break location changes from liquid to steam. (The break "uncovers.")

8. As RCS pressure drops, **break flow** drops and **makeup flow** increases. This trend continues until an equilibrium is reached where break flow equals makeup flow.

9. **Containment pressure** increases rapidly as the hot coolant is released into the containment volume.

Point Explanation

10. **Containment pressure** decreases as the containment fan coolers remove heat from and condense steam in the containment volume. The highest pressure reached in containment (~30 psig) does not quite reach the containment spray initiation setpoint. This can be seen in the RWST level trend. If containment spray had actuated, the rate of RWST depletion would have increased significantly.

11. BIT flow shows sequential starts of the HHSI pumps. The first starts occurs when SI is first initiated. The pumps then trip on loss of offsite power and subsequently start as loads are sequenced on the emergency diesel generators.

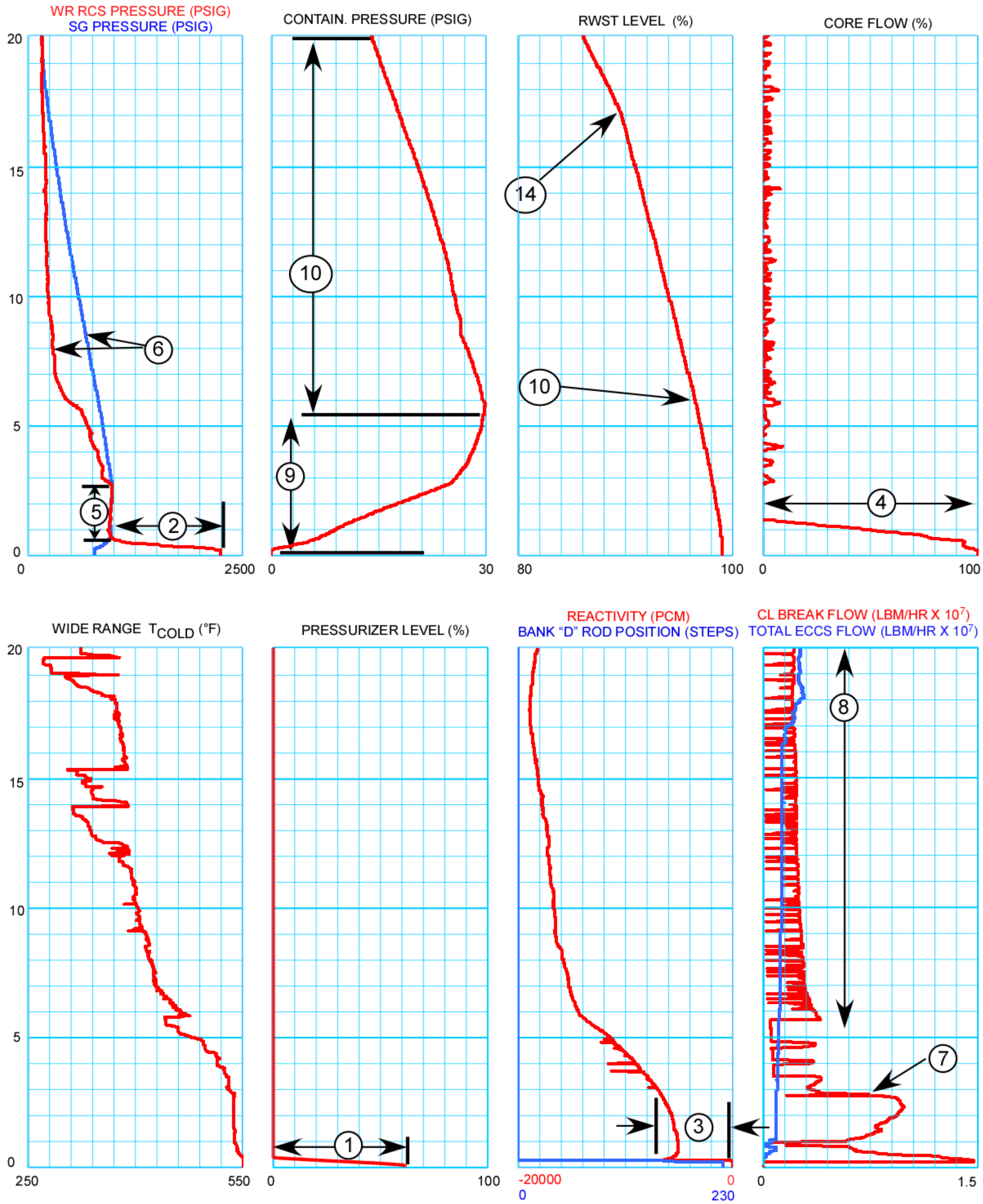
12. **Safety injection system flow** at first increases rapidly with the initiation of intermediate head injection (the safety injection pumps are started by the DBA sequencer a few seconds after the CCPs). This illustrates that the RCS depressurizes quickly to below the shutoff head of the pumps. SI system flow then gradually increases as the pump discharge pressure decreases (characteristic of centrifugal pumps).

13. The drop in **cold-leg accumulator level** reflects the discharge from the accumulators when the RCS pressure falls below the accumulator nitrogen cover pressure (600 psig).

14. **RHR flow** begins when RCS pressure drops below about 190 psig.

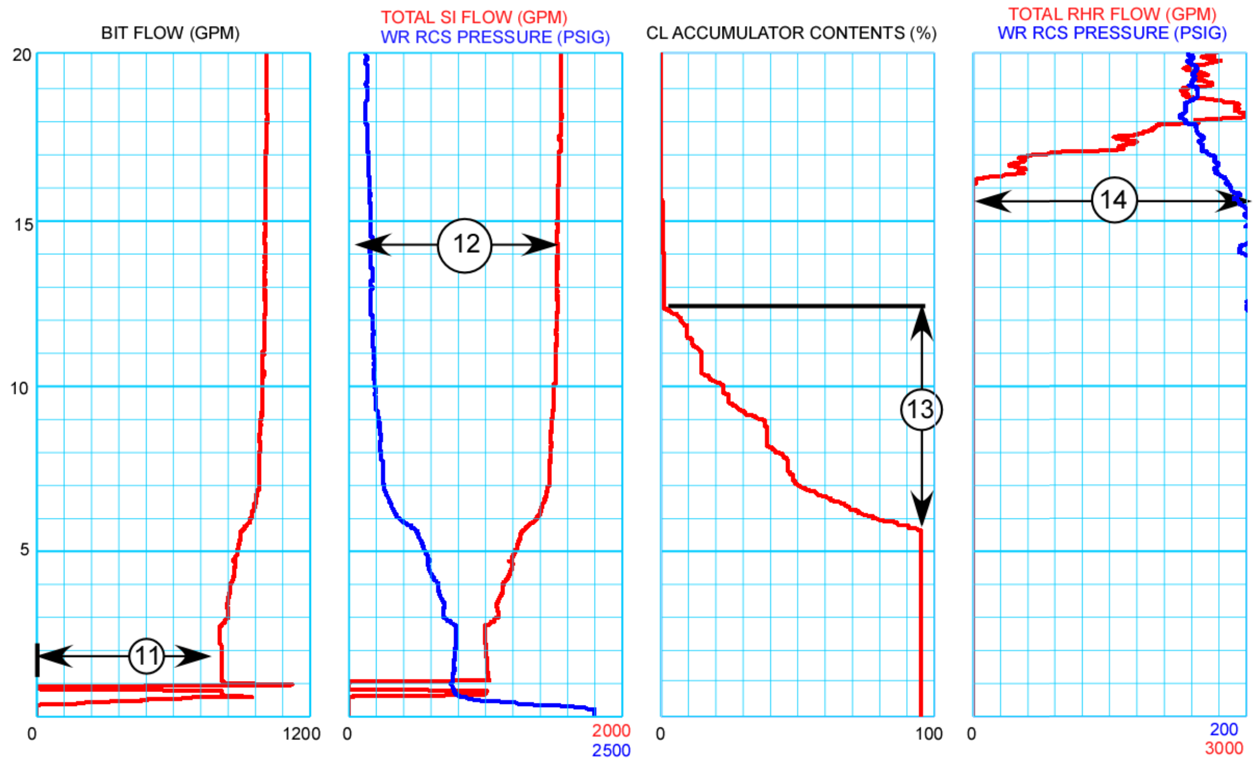
What this transient illustrates:

1. The loss of inventory from the RCS and the reduction in RCS pressure associated with a LOCA, and the resulting protection system responses.
2. For larger RCS breaks, the SGs no longer serve a heat removal function. This condition occurs when RCS pressure drops below SG pressure. At this point, natural circulation stops.
3. The development of saturated conditions in the RCS.
4. The responses of the ECCSs and the different pressures at which they inject.
5. The reduction in containment pressure due to containment fan cooler operation.



Transient 5.76 6-IN. Cold-Leg Break

This page intentionally blank



TRANSIENT 5.76
6 INCH COLD LEG BREAK

Initial Conditions
BOL
Tavg: 585.2°F
Pressurizer Pressure: 2232 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic

Initiating Event:
40,000-gpm (~6" diameter) break in cold leg of loop #1
Failure of fast transfer to offsite power when the main generator trips

Additional Information:

Cold leg break flow	RWST level
BIT flow	Reactivity
Total SI flow	Core flow
Total RHR flow	WR cold-leg temp.
Cold leg accumulator level	
Containment pressure	

Transient 5.76 6-IN. Cold-Leg Break

This page intentionally blank

Transient 5.77 Loss-Of-Feedwater ATWS

Initial Conditions:

BOL	Bank D Rod Position: 219 steps
T_{avg}: 585.3°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	Reactor trip breakers are failed in the closed position
	All control systems in automatic

Initiating Event: Simultaneous trip of both main feed pumps

Point **Explanation**

1. **Feed flow** drops to 0 rapidly with the loss of both main feed pumps.
2. **Generator load** decreases rapidly with the turbine setback (acts through the load limit circuit) initiated by the feed pump trips.
3. **Steam generator level** rapidly decreases out of the narrow-range level indicating range as a result of (1) shrink and (2) the stoppage of feed flow and continued steaming.
4. **Feed flow** comes back on scale with the initiation of AFW on the trip of both main feed pumps.
5. **Generator load** drops to 0 when the turbine is tripped by the ATWS mitigation system actuation circuit (AMSAC). In accordance with the design of the Trojan AMSAC, the trip occurs 25 sec after 3 of the SG levels reach the low-low level setpoint.
6. **T_{avg}** increases with the power mismatch resulting from continued nuclear power generation and the loss of the turbine load. Even the development of full-blast steam dump flow within the first minute cannot halt the increasing trend.
7. **Bank D rod position** decreases as the rod control system inserts the rods at the maximum rate (72 steps/min) in response to large inputs from the power mismatch circuit (setback and turbine trip) and from the temperature mismatch circuit ($T_{avg} >> T_{ref}$).
8. **Nuclear power** decreases rapidly due to the negative reactivity associated with the rod insertion and the coolant temperature increase.
9. **Steam dump demand** rapidly increases to maximum with the large T_{avg} - T_{ref} difference. The steam flow graph indicates that the dumps are first armed by a loss-of-load arming signal and then remain armed by a turbine trip arming signal following the turbine trip.
10. **Steam flow** and **steam pressure** decrease as the SG inventories are boiled off to near empty.
11. The increase in **T_{avg}** accelerates as the SG heat sink is almost completely gone.

Point Explanation

12. **Wide-range RCS pressure** reflects PORV and safety valve lifts as the pressurizer is solid and coolant temperature is still increasing. At least some of the valve lifts probably result in water release.

13. **Feed flow** increases when the SG pressure has dropped below the condensate pump discharge pressure.

14. **T_{avg}** decreases rapidly with the abrupt increase in SG heat transfer associated with the introduction of condensate flow. At this point the core heat output has been reduced to decay heat levels by the rod insertion.

15. **Pressurizer level** drops rapidly with the rapid contraction of the reactor coolant associated with the coolant temperature decrease.

16. **Wide-range RCS pressure** rapidly drops to below the low pressure ESF actuation setpoint (1807 psig) with the expansion of the steam bubble accompanying the coolant contraction.

17. The rapid increase in **BIT flow** (followed closely by the rapid increase in **safety injection system flow**) reflects the ESF actuation and the low RCS pressure.

18. **Pressurizer level** increases as ECCS flow (with letdown isolated) fills the pressurizer.

19. **Wide-range RCS pressure** increases to the PORV lift setpoint as the pressurizer fills again due to ECCS flow.

20. **Wide-range RCS pressure** "squiggles" reflect PORV lifts after the pressurizer goes solid. Instrument air to containment has been isolated on the ESF actuation, but the pressurizer PORVs are equipped with air accumulators.

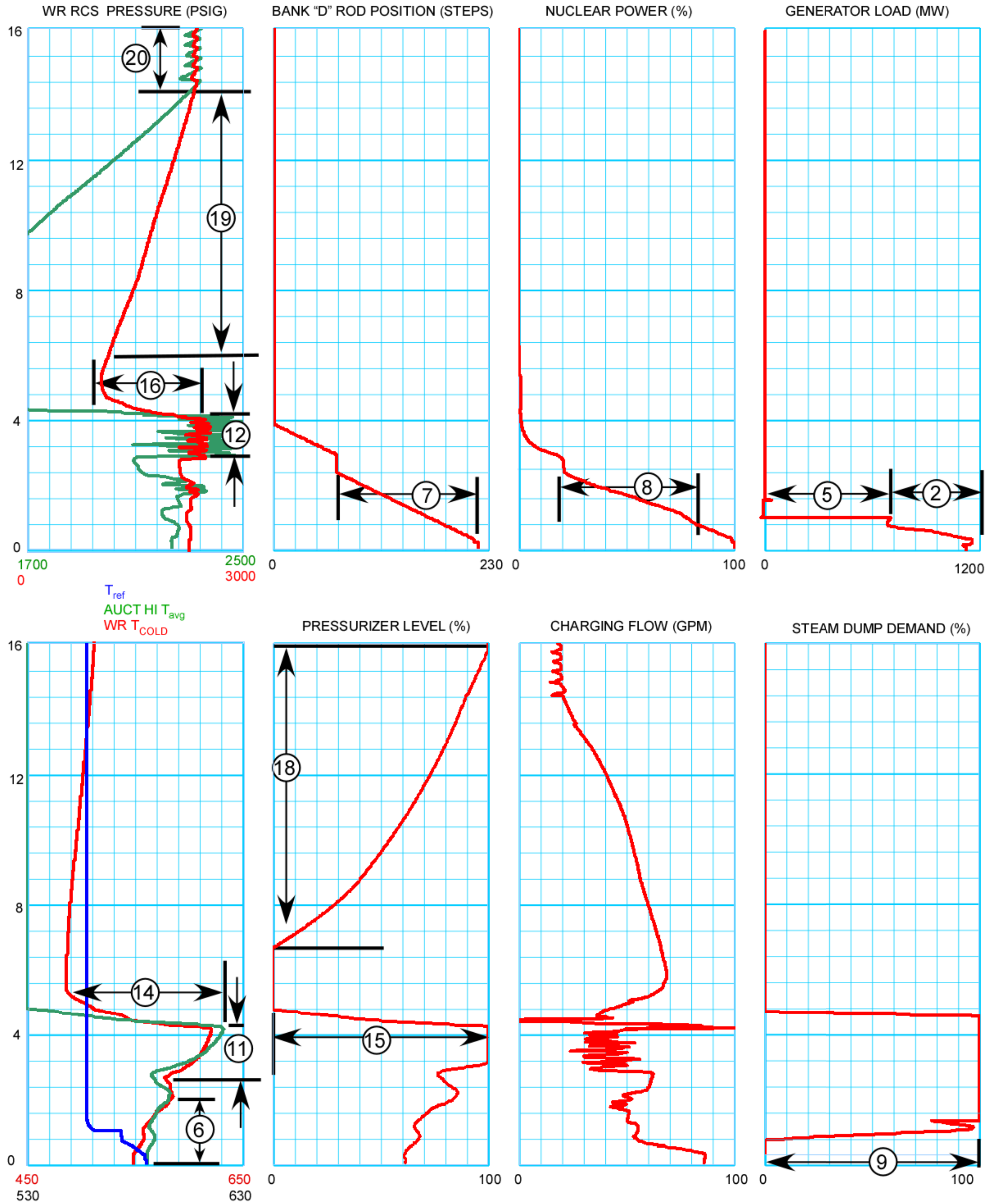
21. Negative **reactivity** is added by control rod insertion and RCS heatup.

22. Positive **reactivity** is added by RCS cooldown. The moderator temperature coefficient is about -8 pcm/% during this interval, which is a typical BOL value.

23. Negative **reactivity** is added by ECCS flow.

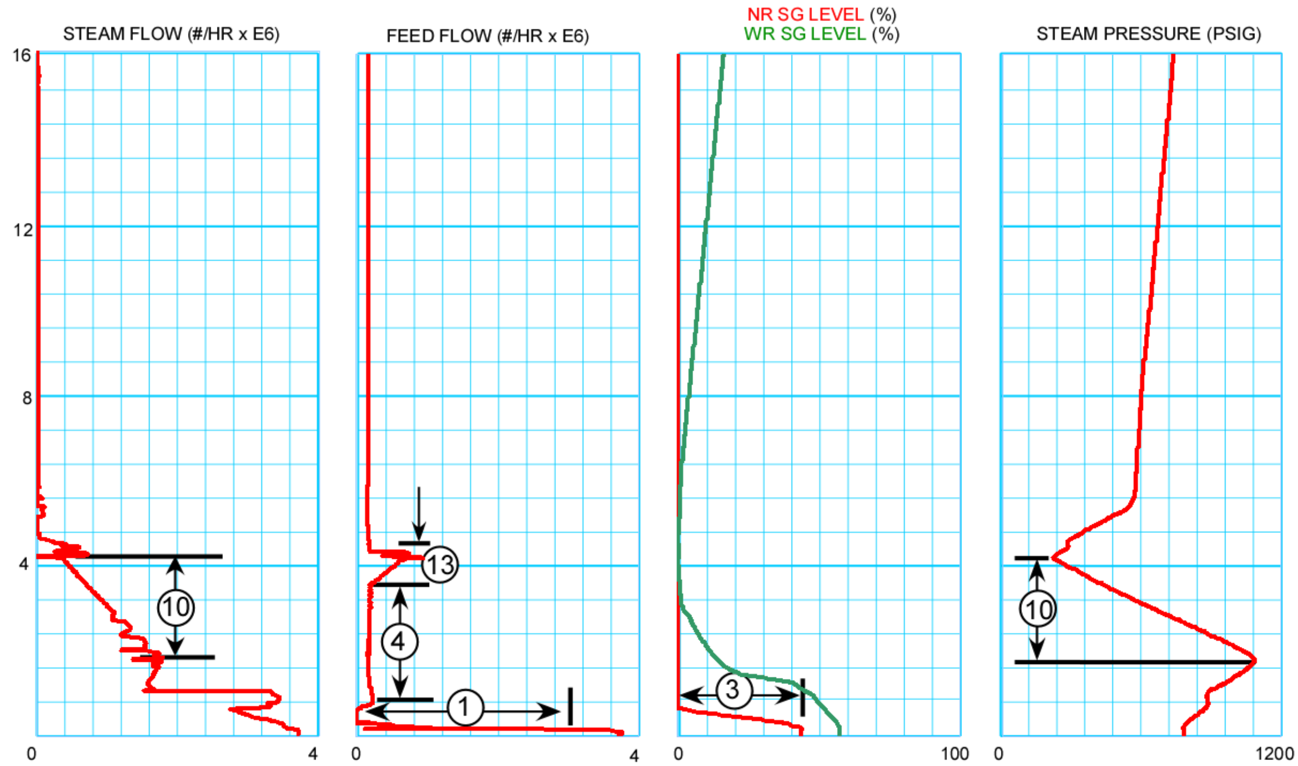
What this transient illustrates:

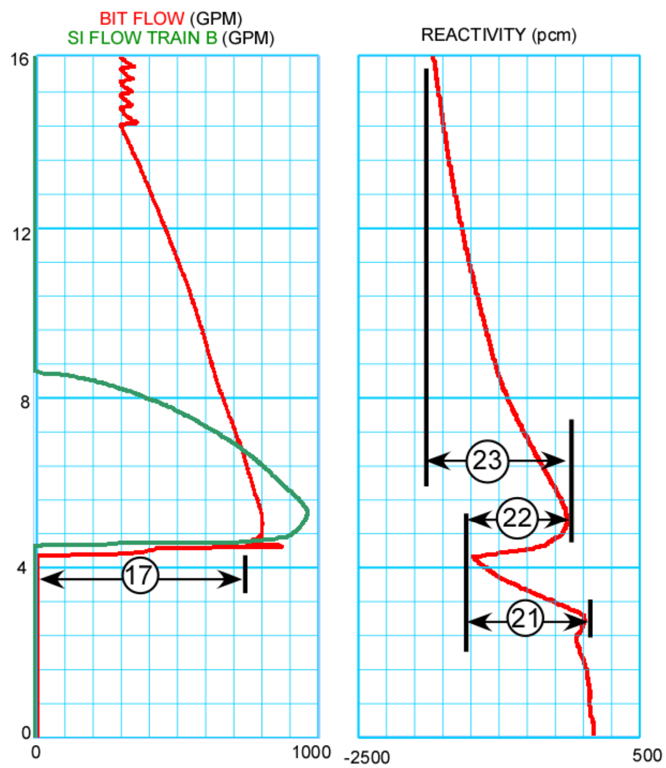
1. The large reduction in heat removal capability with the loss of main feedwater, and the inability of AFW alone to hold T_{avg} at normal operating values at high reactor powers.
2. An ESF actuation and the responses of ESF systems.
3. The development of solid plant conditions due to overheating and overfilling of the RCS.
4. A turbine setback caused by the loss of a main feed pump.
5. The reactor achieves and maintains subcriticality even with the failure of the reactor protection system and no operator action. The safety significance of this transient is the challenge to RCS integrity due to the severe overpressure transient during the initial RCS heatup. Fuel integrity is not challenged in this event.



Transient 5.77 Loss-Of-Feedwater ATWS

This page intentionally blank





TRANSIENT 5.77
LOSS OF FEED ATWS - BOL

Initial Conditions
BOL
Tavg: 585.3°F
Pressurizer Pressure: 2230 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic
Reactor trip breakers are failed in the closed position

Initiating Event:
Both main feed pumps trip simultaneously

Additional Information:
Wide range steam generator level
BIT flow
SI flow train B
Reactivity

Transient 5.77 Loss-Of-Feedwater ATWS

This page intentionally blank

Transient 5.78 Loss-Of-Feedwater ATWS - EOL

Initial Conditions:

EOL	Bank D Rod Position: 219 steps
T_{avg}: 585.3°F	Nuclear Power: 100%
PZR Pressure: 2230 psig	Reactor trip breakers are failed in the closed position
	All control systems in automatic

Initiating Event: Simultaneous trip of both main feed pumps

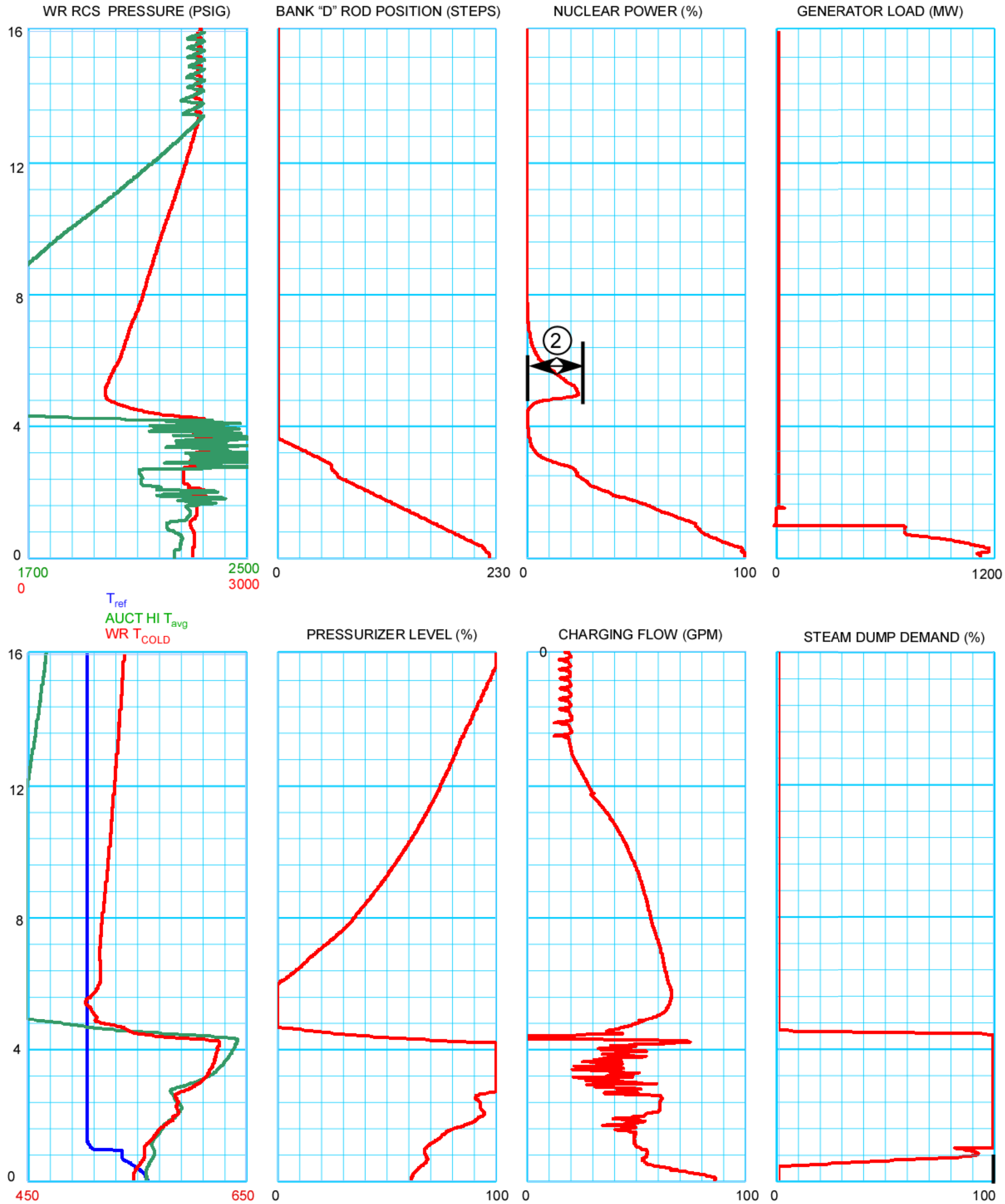
<u>Point</u>   <u>Explanation</u>

1. **Feed flow** drops to 0 rapidly with the loss of both main feed pumps.
2. The RCS cooldown brings the reactor supercritical. The EOL moderator temperature coefficient is larger in magnitude than the BOL moderator temperature coefficient. Reactor power exceeds 20% until ECCS flow makes the reactor subcritical again.
3. Negative **reactivity** is added by ECCS flow.

What this transient illustrates:

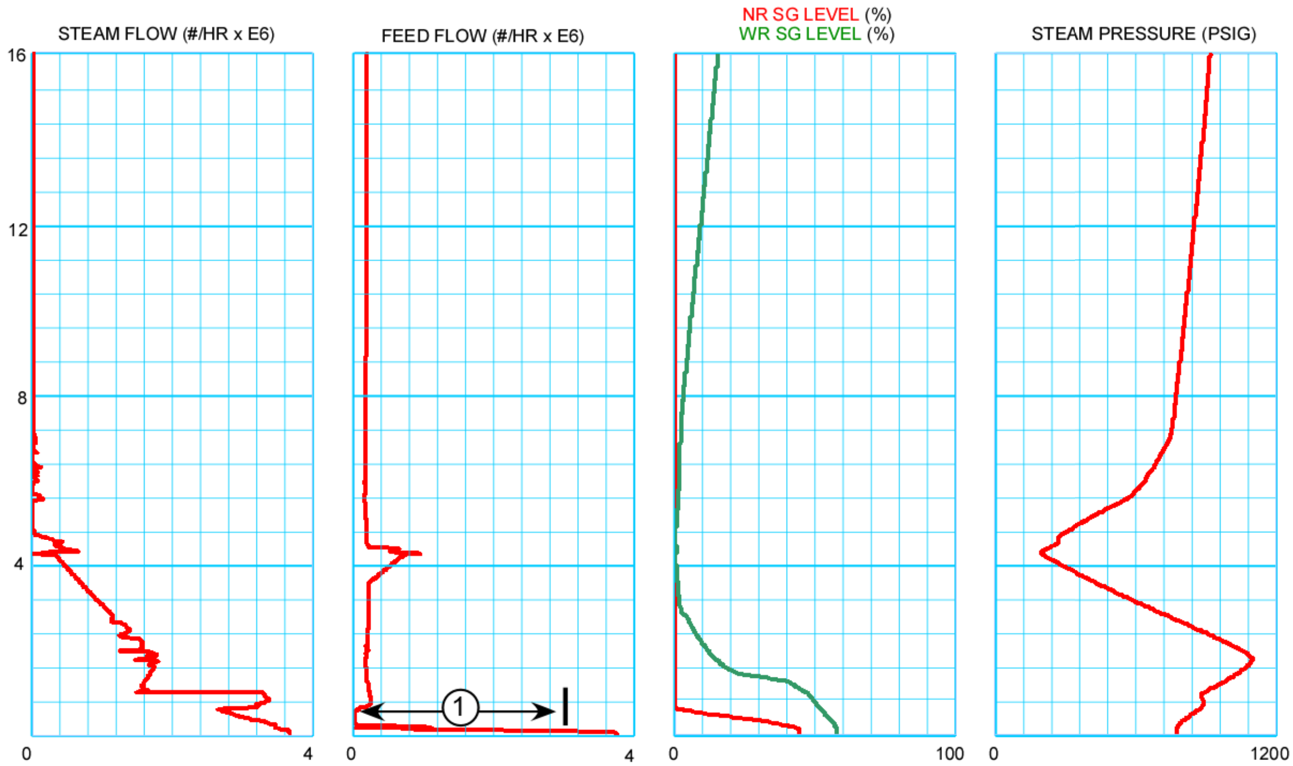
1. When compared to BOL conditions (Transient 5.77), the EOL moderator temperature coefficient is larger in magnitude, which makes the reactivity effect of a cooldown more significant.
2. The reactor returns to criticality, but ECCS flow achieves and maintains subcriticality.

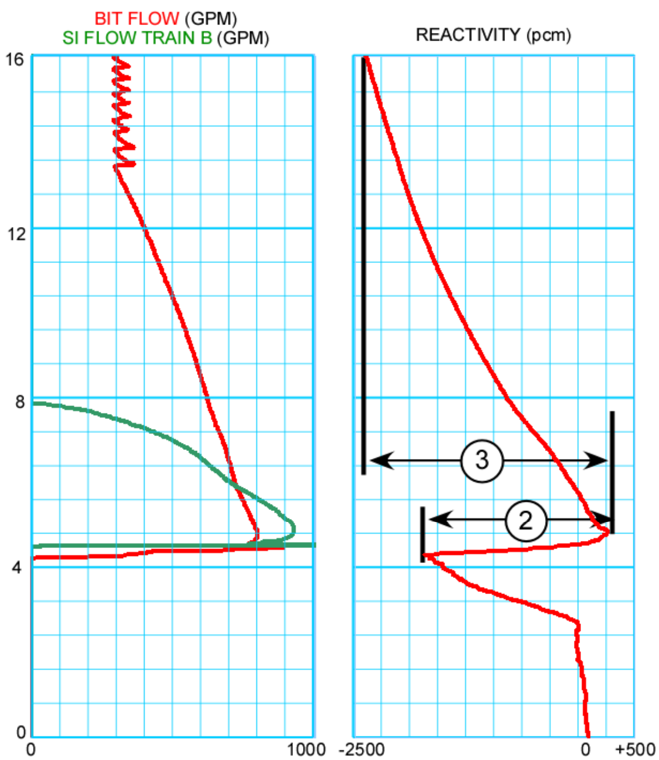
This page intentionally blank



Transient 5.78 Loss-Of-Feedwater ATWS - EOL

This page intentionally blank





TRANSIENT 5.78
LOSS OF FEED ATWS - EOL

Initial Conditions
EOL
Tavg: 585.3°F
Pressurizer Pressure: 2230 psig
Nuclear Power: 100%
Bank D Rod Position: 219 steps
Charging flow provided by one centrifugal charging pump
All control systems in automatic
Reactor trip breakers are failed in the closed position

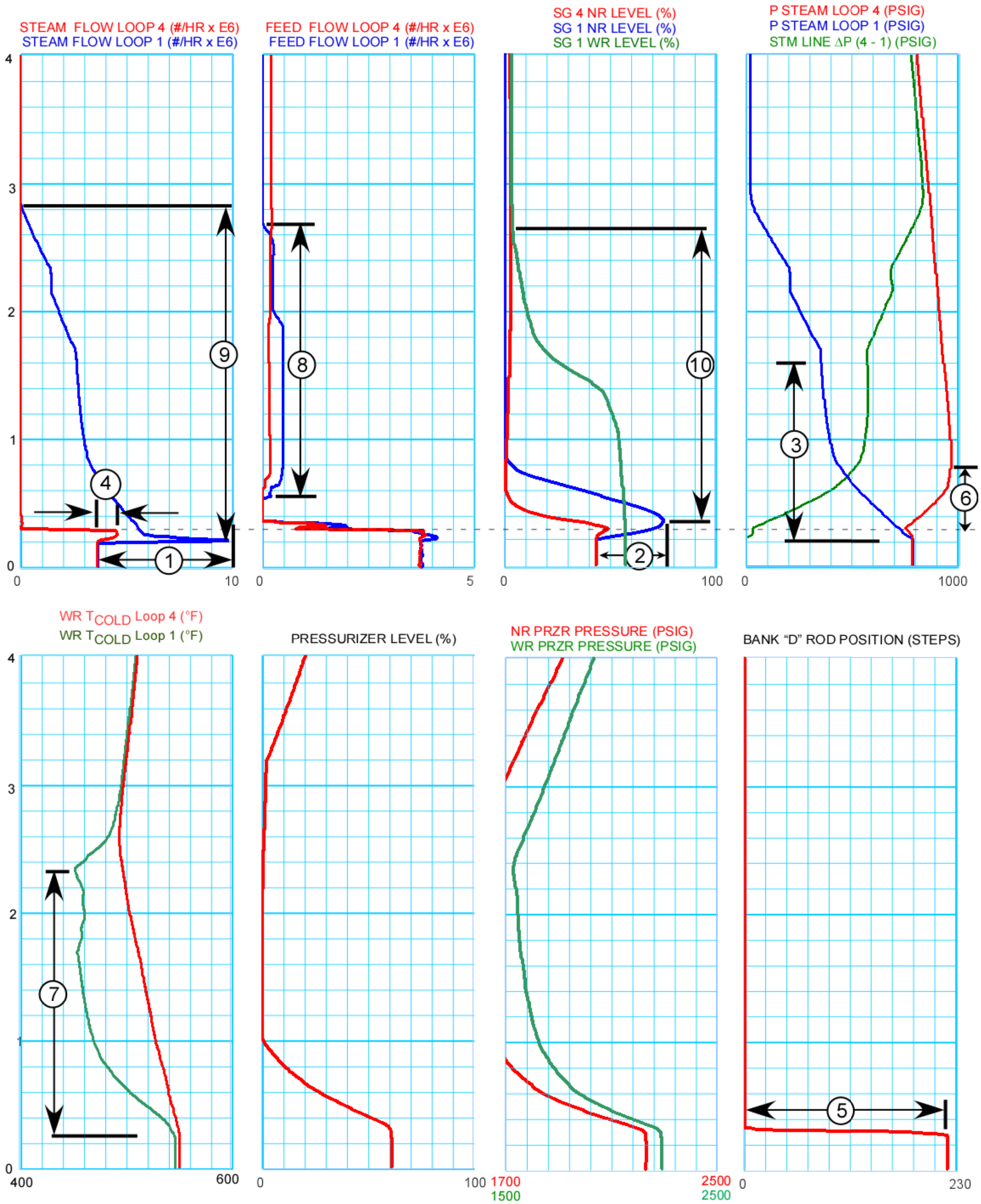
Initiating Event:
Both main feed pumps trip simultaneously

Additional Information:
Wide range steam generator level
BIT flow
SI flow train B
Reactivity

Transient 5.78 Loss-Of-Feedwater ATWS - EOL

This page intentionally blank

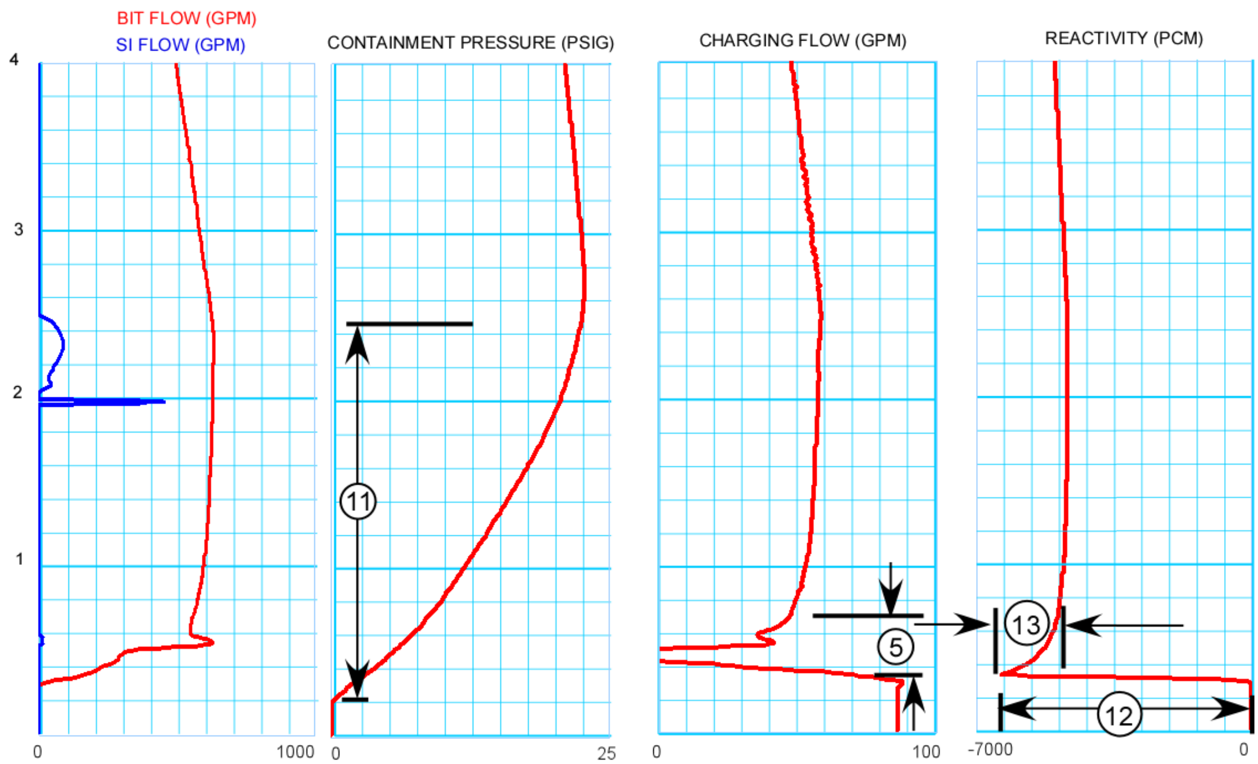
Exercise 1

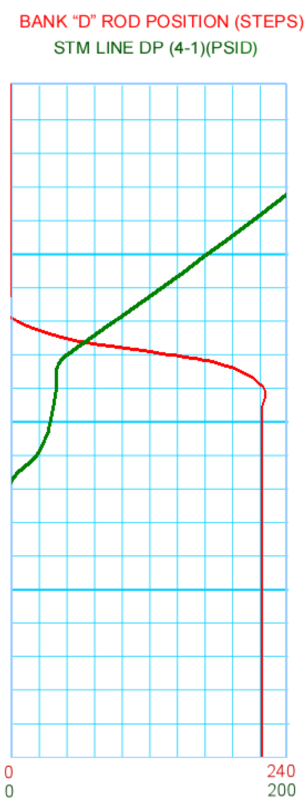


Exercise #1

This page intentionally blank

Exercise 1 (cont.)





What was the initiating event?

What signal caused the reactor trip?

What evidence supports your conclusion in the previous question?

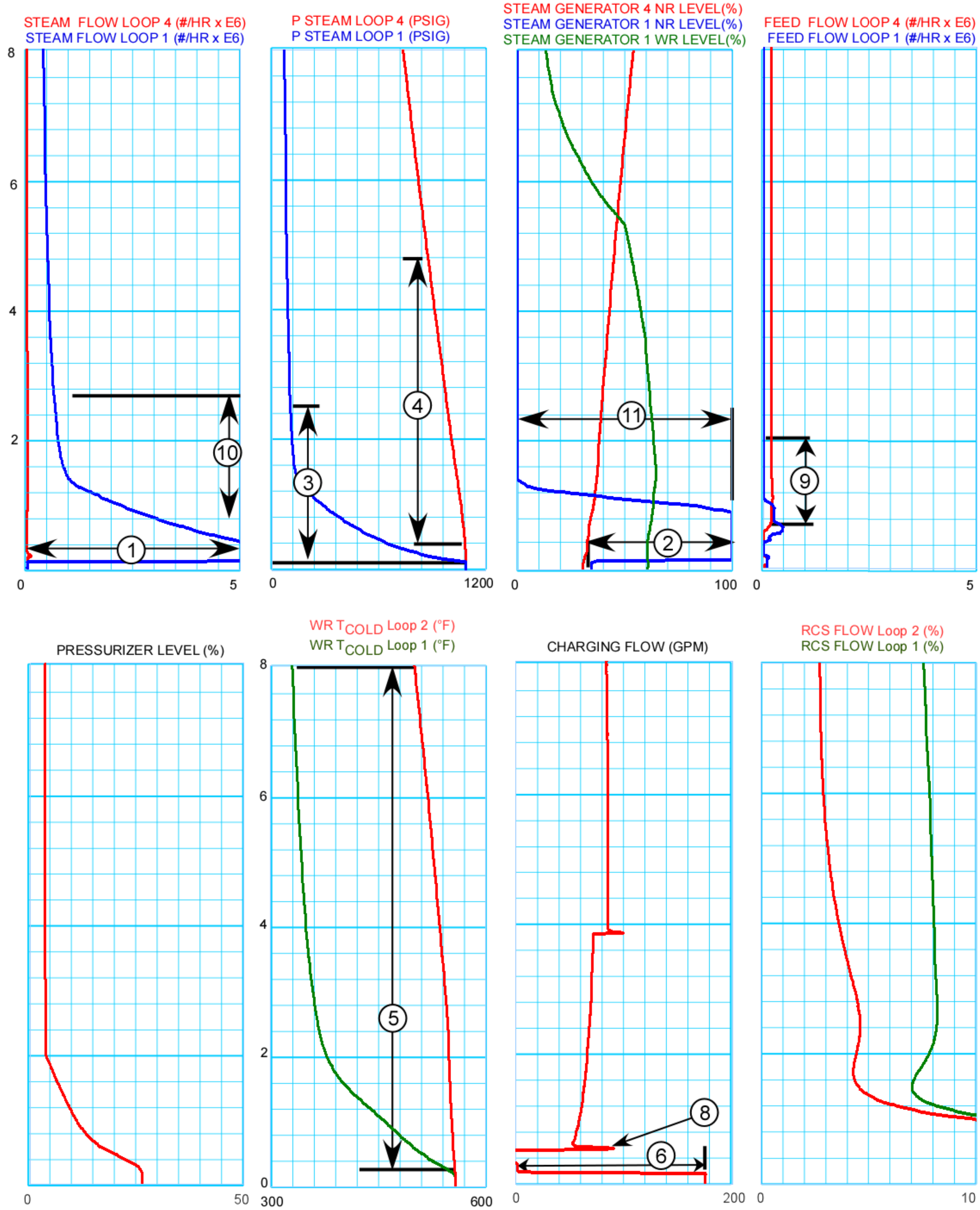
Loop 4 Tcold has a minimum value of about 496°F, which corresponds to a saturation pressure of 656 psia. Describe the heat transfer in the loop 4 SG.

Explain the trend at each numbered point.

Exercise #1

This page intentionally blank

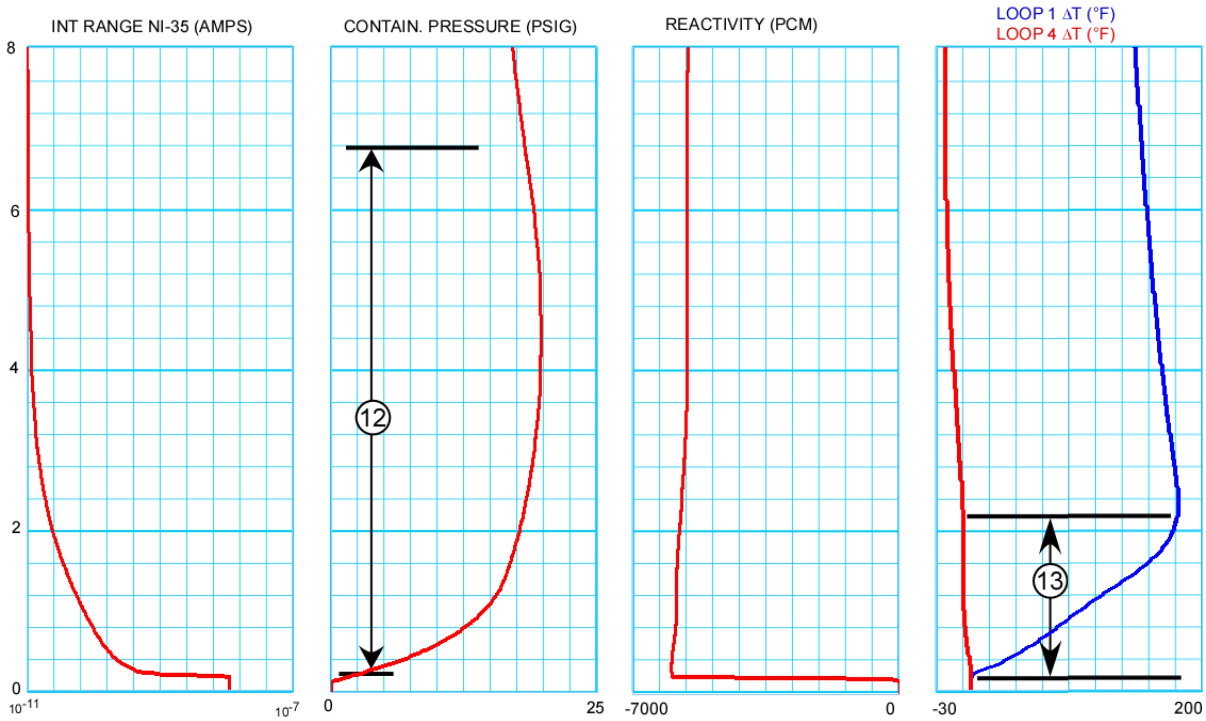
Exercise 2

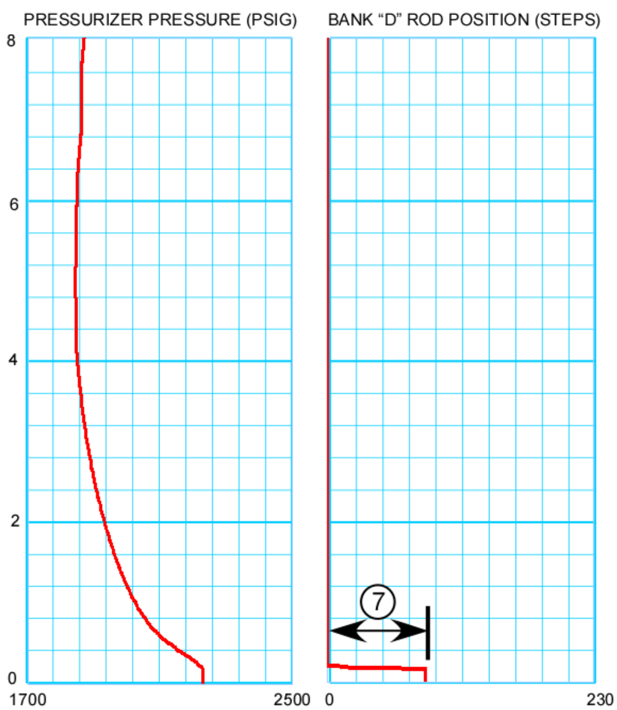


Exercise #2

This page intentionally blank

Exercise 2 (cont.)





How is the initiating event different than in Exercise #1?

Why is the cooldown so much larger than in Exercise #1?

Compare the elapsed time with Exercise #1. Why does this event progress so slowly?

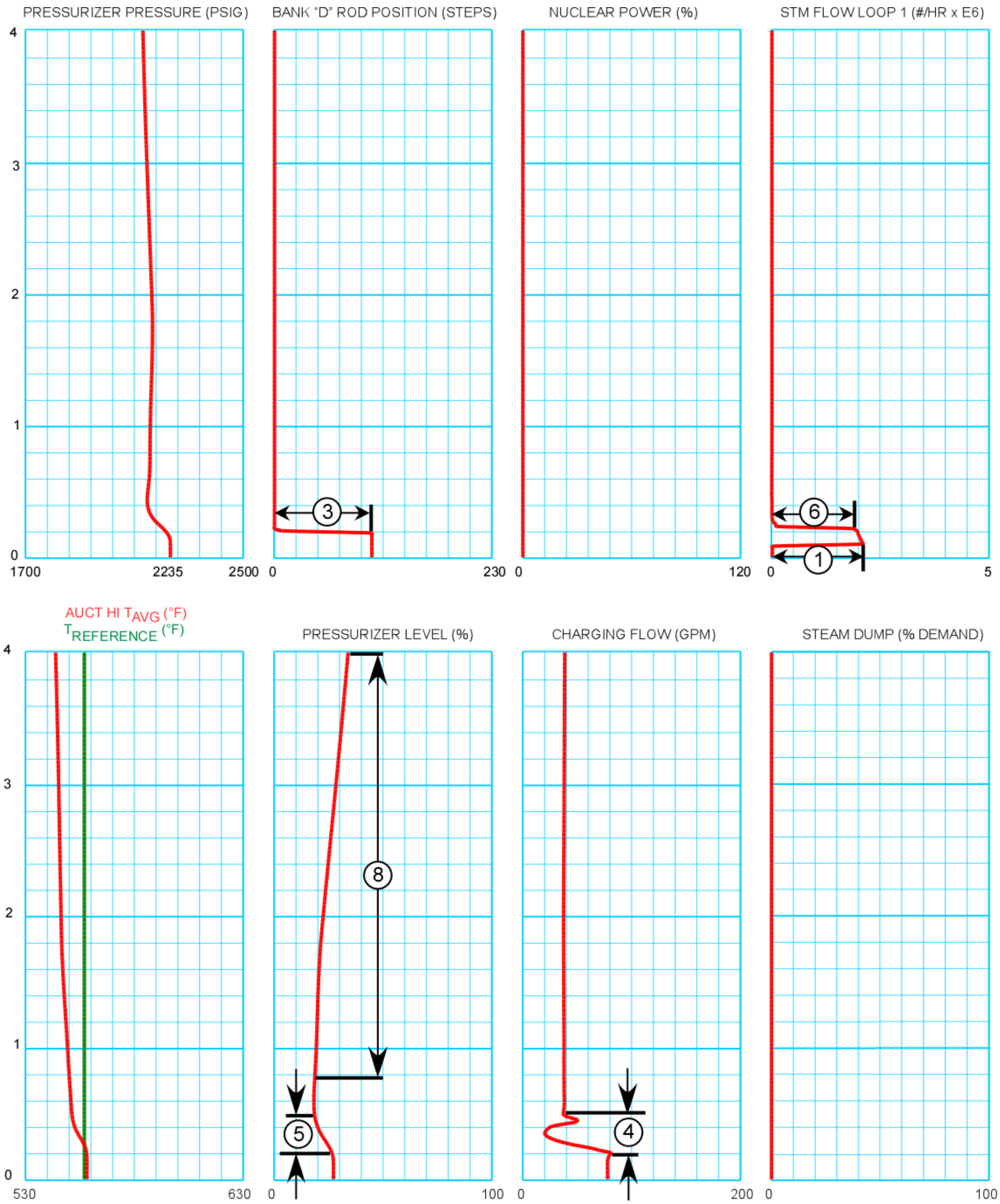
Compare and explain the differences in ESFAS signals that actuate in Exercises #1 and #2.

Explain the trends at the numbered points.

Exercise #2

This page intentionally blank

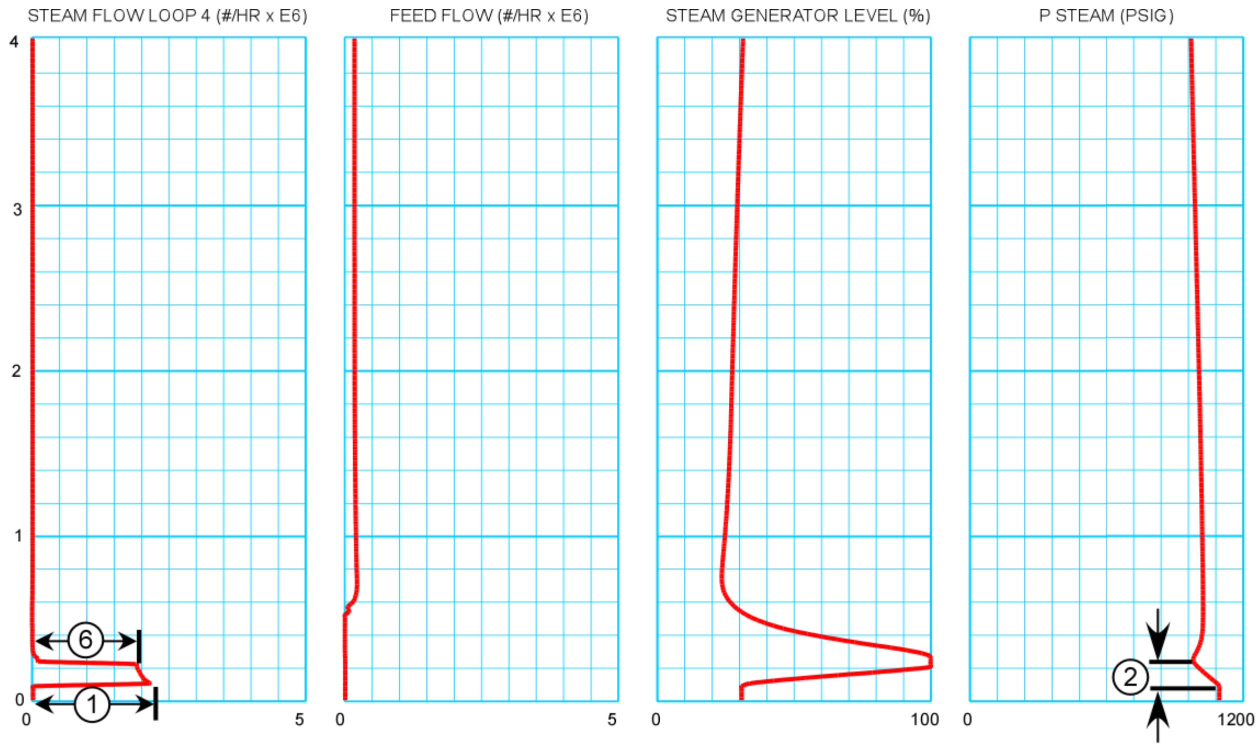
Exercise 3

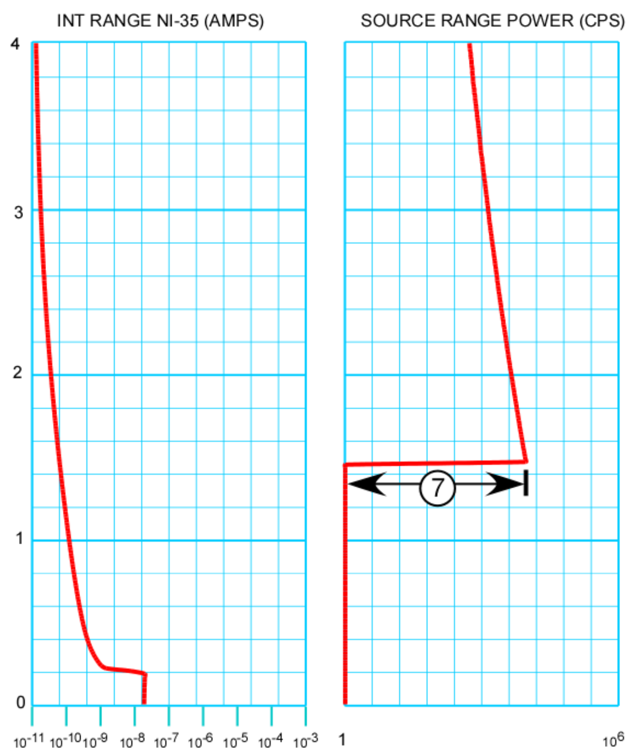


Exercise #3

This page intentionally blank

Exercise 3 (cont.)





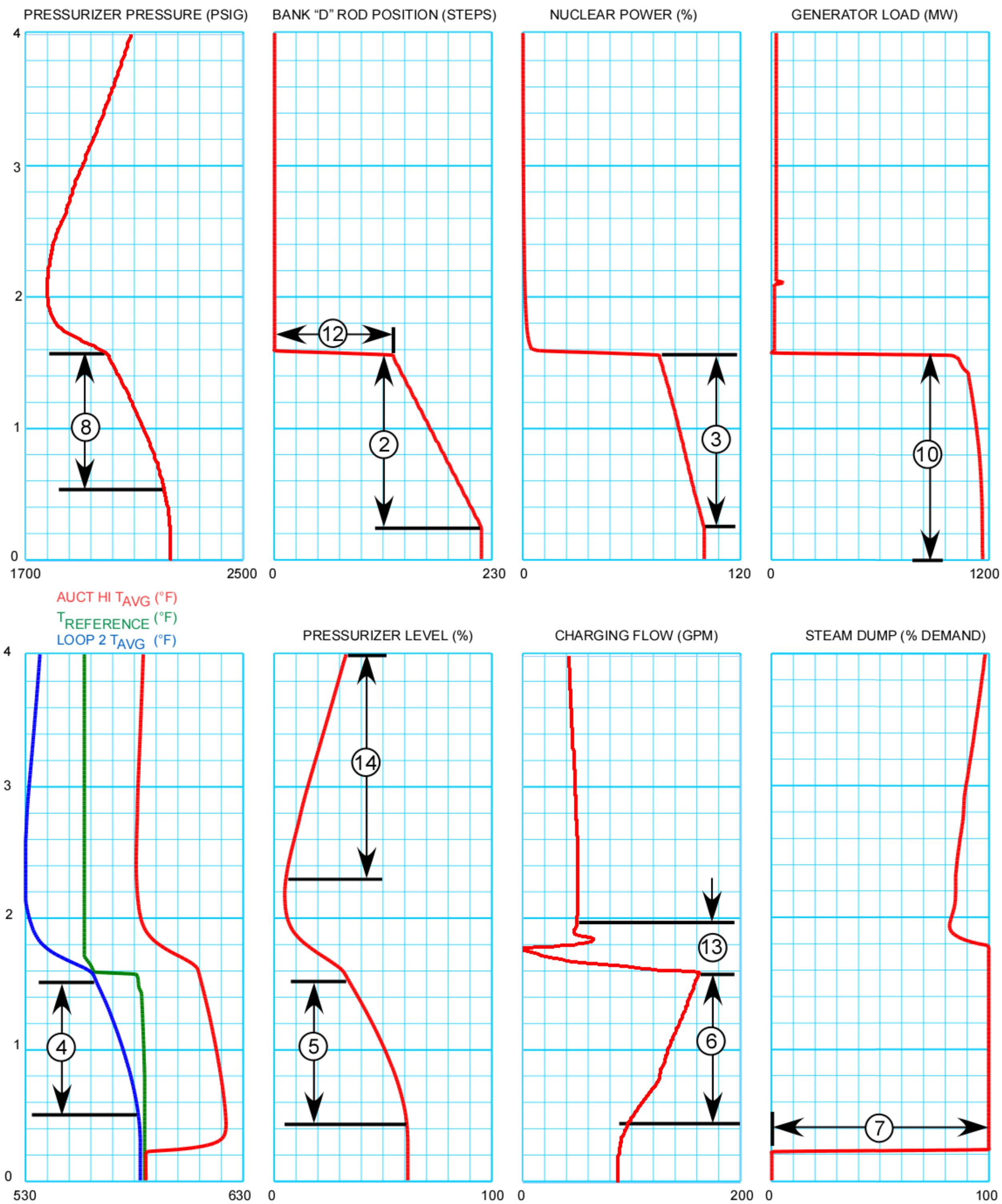
What is the initiating event?

Explain the trends at the numbered points.

Exercise #3

This page intentionally blank

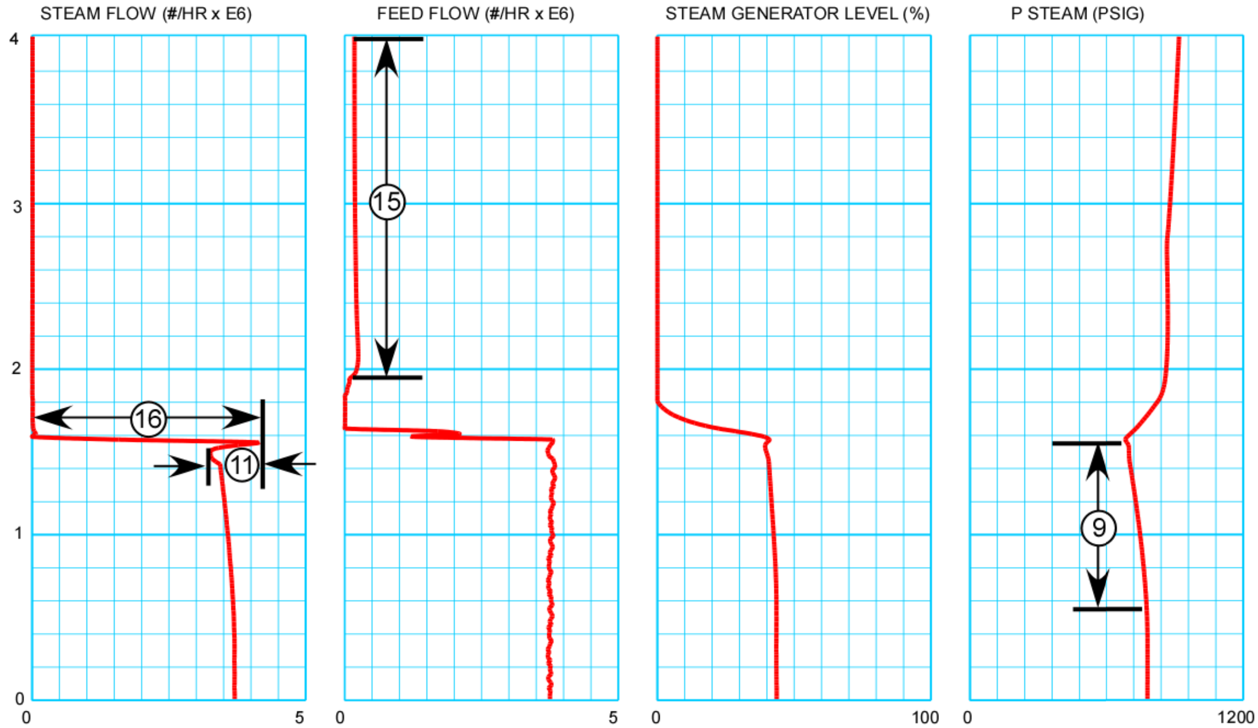
Exercise 4



Exercise #4

This page intentionally blank

Exercise 4 (cont.)

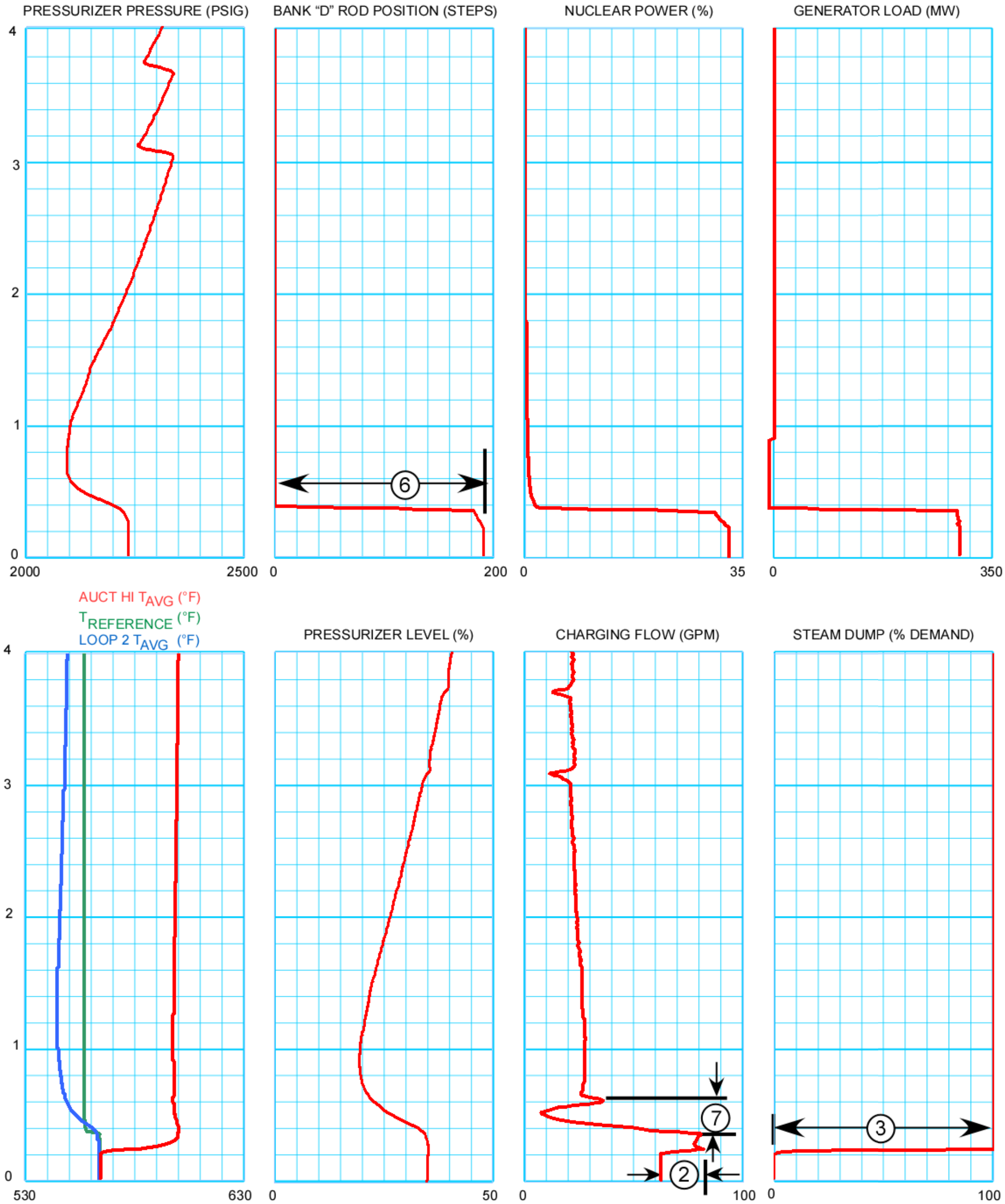


1. What was the initiating event?
2. What was the cause of the reactor trip?
3. Explain the trend at each numbered point.

Exercise #4

This page intentionally blank

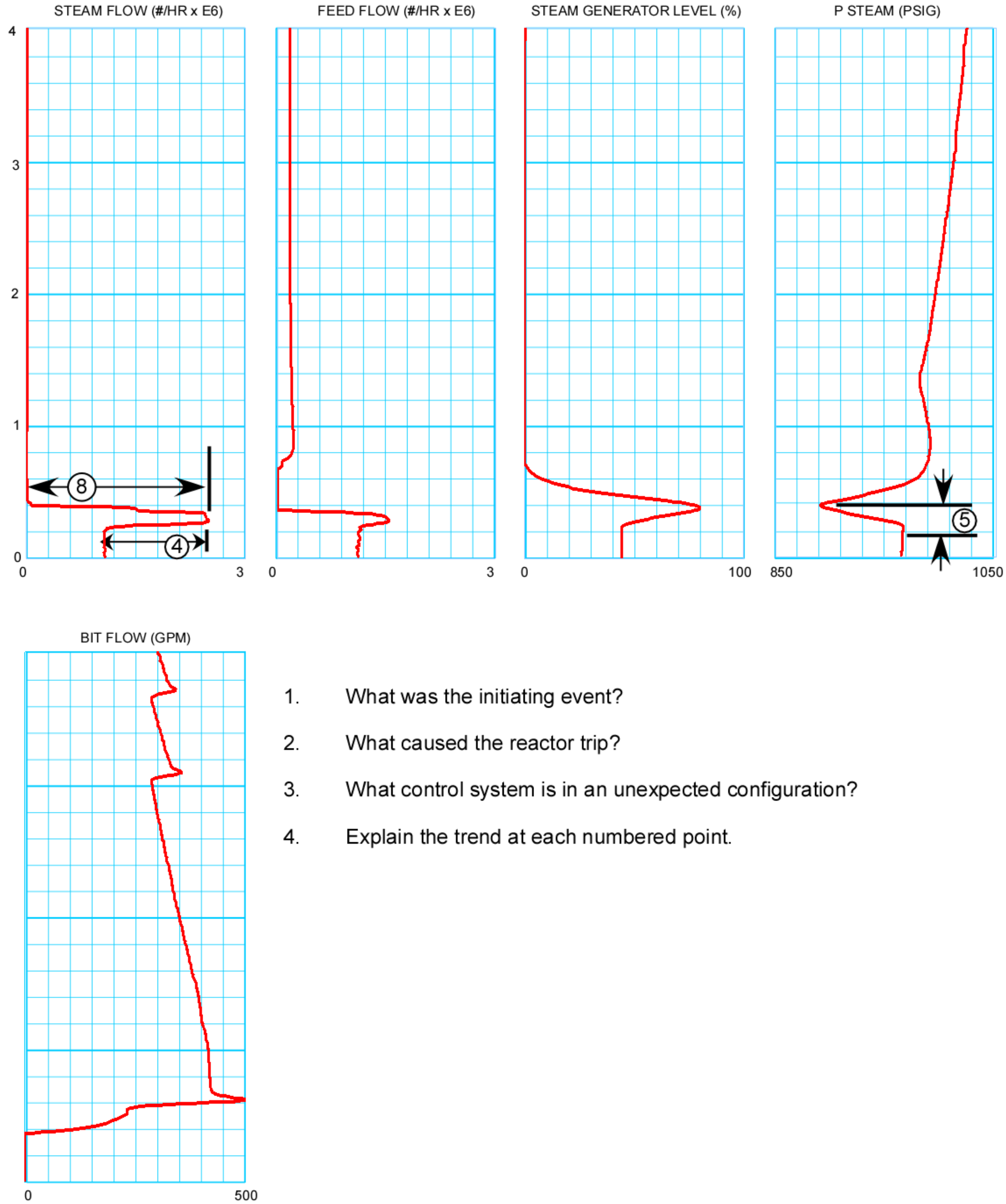
Exercise 5



Exercise #5

This page intentionally blank

Exercise 5 (cont.)

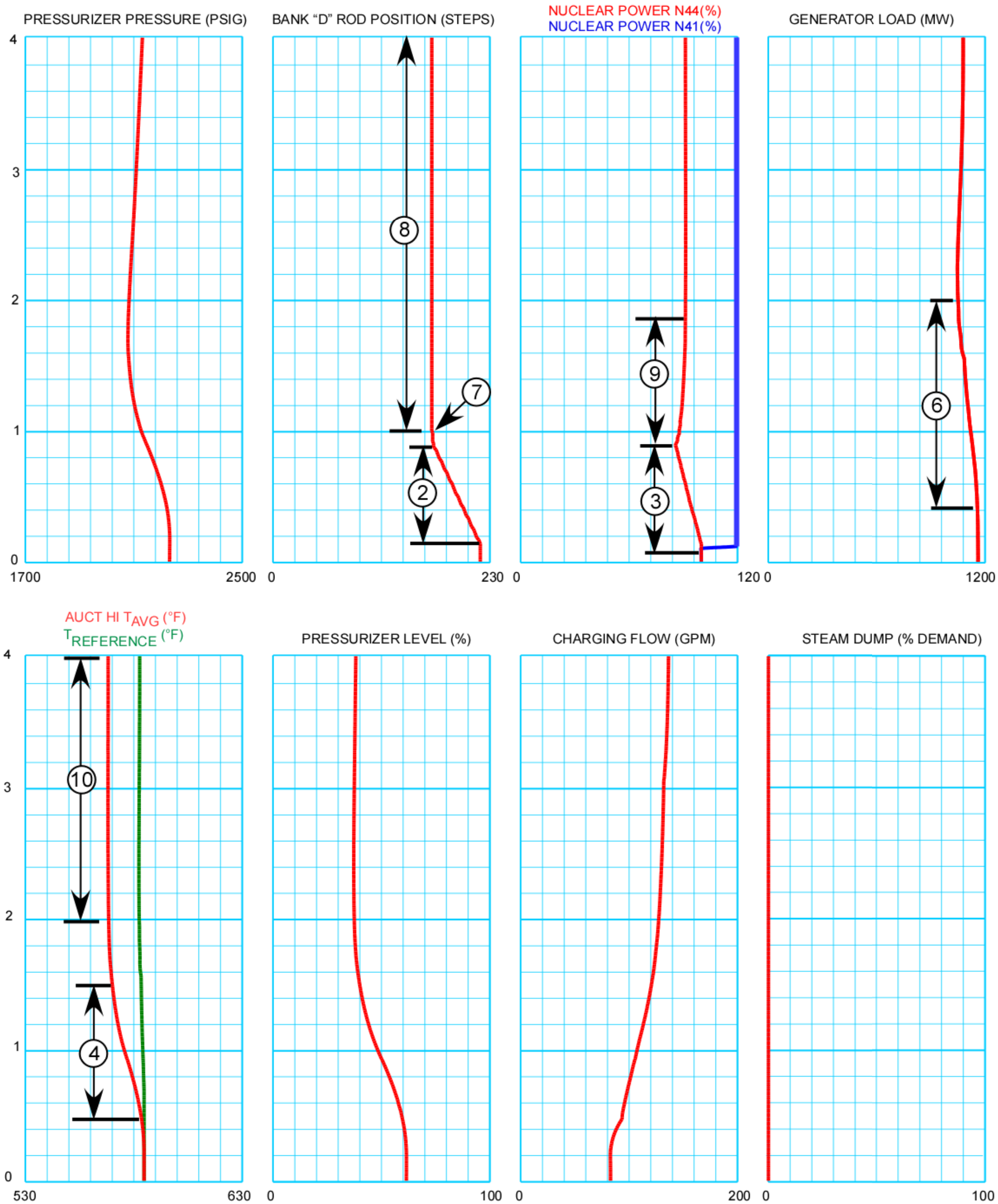


1. What was the initiating event?
2. What caused the reactor trip?
3. What control system is in an unexpected configuration?
4. Explain the trend at each numbered point.

Exercise #5

This page intentionally blank

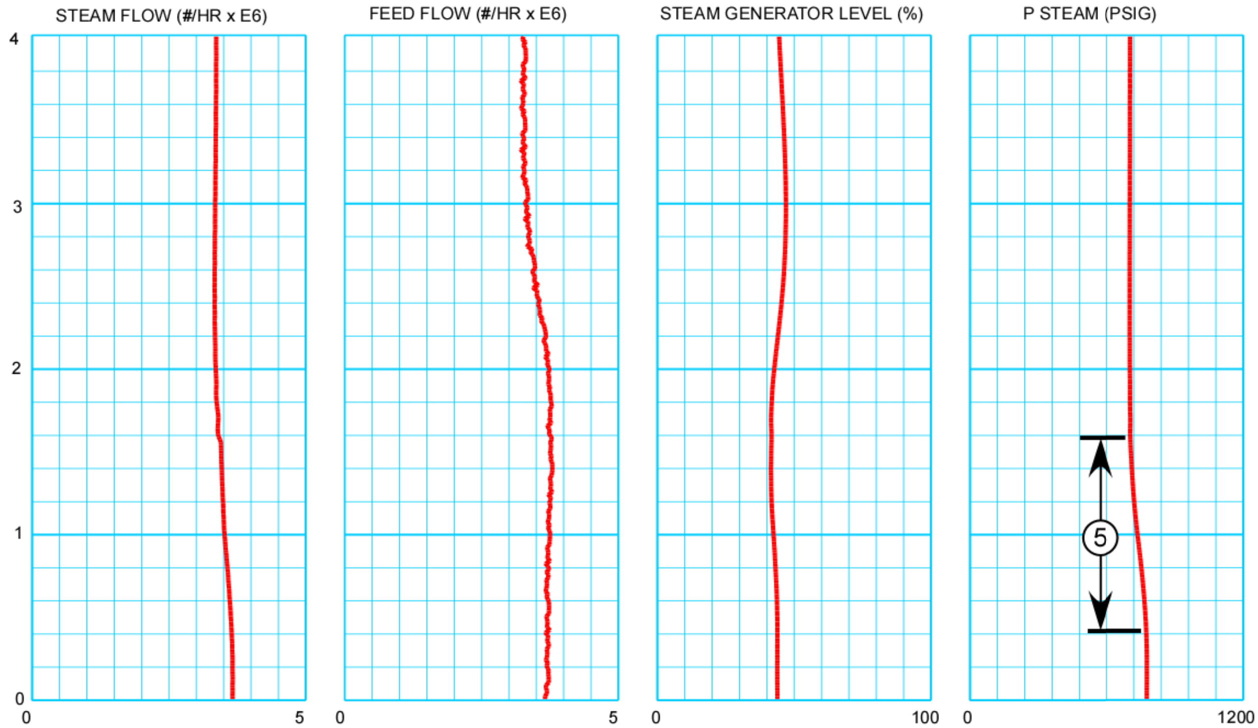
Exercise 6



Exercise #6

This page intentionally blank

Exercise 6 (cont.)



1. What was the initiating event?

2. Explain the trend at each numbered point.

Exercise #6

This page intentionally blank